1 | Origin: New | ☑ R | ☐ SRO | ☐ Student Handout? | ☑ Lower Order? | Question ID: 6000023
☑ Selected for Exam | ☐ Past NRC Exam?

The plant has been tripped from 30% power per AOP-2575, Rapid Downpower. The Crew has since transitioned to EOP 2526, Reactor Trip Recovery and is performing applicable actions.

Which of the following plant conditions, taking each choice SEPERATELY, would REQUIRE the use of an AOP, in conjunction with the use of EOP-2526?

A The plant shutdown was required by AOP-2560, Storms, High Winds and High Tides, due to an approaching hurricane. ☑

B The trip from 30% power was required by AOP-2575, Rapid Downpower, due to abnormal ASI distribution. ☐

C The plant shutdown was required by AOP-2556, CEA Malfunctions, due to a dropped CEA that could not be recovered. ☐

D While performing EOP-2526, the CONVEX must deenergize the RSST due to an internal fault in the transformer. ☐

Justification

A - CORRECT: AOP-2560 has further guidance for preparing the plant for an approaching storm that is not covered in any other Operations procedure. Shutting down the plant is just one of the many preparatory actions that must be accomplished before the storm actually arrives.

B - WRONG: If ASI cannot be controlled due to the inability to move CEAs, then AOP-2575 has guidance to trip the reactor when ASI gets too close to the LPD setpoint. Once the plant is tripped, ASI is irrelevant.

C - WRONG: Once the plant is tripped, all CEAs are fully inserted. Therefore AOP-2556 would no longer be of use.

D - WRONG: If the RSST is lost while performing EOP-2526 the crew must transition to EOP-2528, Loss Of Forced Circulation. The AOP dealing electrical problems does not cover the required actions for Natural Circulation.

References

OP 2260, Unit 2 EOP User's Guide, Rev. 009-00,
AOP 2575, Rapid Downpower, Rev. 003-05

Comments and Question Modification History

NRC K/A System/E/A

System E02 Reactor Trip Recovery

Number EK3.2 **RO** 2.8 **SRO** 3.5 **CFR Link** (CFR: 41.5 / 41.10, 45.6, 45.13)

Knowledge of the reasons for normal, abnormal and emergency operating procedures associated with Reactor Trip Recovery as they apply to the Reactor Trip Recovery.

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

# 1	Origin: Org	☑ R ☐ Selected for Exam	☐ SRO	☐ Student Handout? ☐ Past NRC Exam?	☐ Lower Order?	Question ID: 6000000

The plant has tripped from 100% power due to grid instabilities, which have since been resolved. The Crew has transitioned to EOP 2526, Reactor Trip Recovery and is performing applicable actions.

Then, the crew receives results from Chemistry of the RCS boron concentration and recognize that they soon will not meet Shut Down Margin requirements.

In accordance with plant procedure usage requirements, which of the following describe the reason why AOP-2558, Emergency Boration, could be utilized by the Crew to raise the RCS Boron concentration at this time?

A The AOP takes into consideration post-trip conditions like EOP 2541, Standard Appendices. ☑

B The AOP specific actions are more inclusive than similar steps in any Event or Functional EOP. ☐

C All AOPs are designed for the unusual system alignments that ESAS actuation could cause. ☐

D All AOPs are written by Subject Matter Experts or System Engineers and then SORC approved. ☐

Justification

A - CORRECT: OP 2260 states that the Functional Recovery procedures (EOP 2540 series) might be entered from an ORP (Optimal Recovery Procedure) if an ORP had been initially selected, but was subsequently found to be inadequate. In this case, Boric Acid flow is degraded, but the safety function is still being met; therefore, the Function Recovery Procedure is NOT appropriate. OP 2260 also states that other procedures, NOPs, AOPs, ARPs) may be used as directed by the US.

B - WRONG: AOPs may be utilized during any EOP if they do not conflict with the EOP guidance, as they are not developed for accident conditions.

C - WRONG: NOPs may be used to supplement EOP usage; however, the EOP is the controlling document because the Ops are NOT validated for accident conditions.

D - WRONG: Although Training Materials ARE developed and validated by personnel more technically qualified on the applicable system than SORC members, SORC members take into account potential plant conditions and administrative requirements beyond the narrow focus of the System Descriptions.

References

OP 2260, Unit 2 EOP User's Guide, Rev. 009-00, Step 1.3.3.b and 1.9.3.b. and EOP

Comments and Question Modification History

Distractor B and D are inplausible.
[re-wrote "B", "C" and "D" to improve plausibility of distractors - RLC]
*** "B" may be too close to correct ***
{Generated new question for the K/A - RLC}

NRC K/A System/E/A

System E02 Reactor Trip Recovery

Number EK3.2 **RO** 2.8 **SRO** 3.5 **CFR Link** (CFR: 41.5 / 41.10, 45.6, 45.13)

Knowledge of the reasons for normal, abnormal and emergency operating procedures associated with Reactor Trip Recovery as they apply to the Reactor Trip Recovery.

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

2 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Mod ☑ Selected for Exam ☐ Past NRC Exam? **6054015**

The plant has tripped from 100% power due to a Main Turbine trip. The following conditions now exist:

- A PORV is stuck open on the trip and its block valve is deenergized.
- RCS pressure has stabilized at ~1200 psia.
- RVLMS has just dropped to 29%.
- Pressurizer level is 60% and rising.
- Both Steam Generators are at 900 psia and stable.
- The "C" HPSI pump has tripped on overload.

Which one of the following actions is required and sufficient to ensure adequate core cooling?

A The open PORV Block Valve must be re-energized and closed. ☐

B Start the "B" HPSI pump, aligned to Fac. 2, to restore HPSI capacity. ☐

C All available Charging Pumps must be running with Letdown secured. ☐

D RCS pressure must be reduced by increasing RCS heat removal. ☑

Justification

D - CORRECT: This event is a small break LOCA. A cooldown will result in a decrease in RCS pressure allowing HPSI to inject.

A - WRONG: While this will stop the LOCA (loss of RCS inventory), it will likely take some time to re-energize the bus and close the block valve. The time it takes to re-energize the block valve may be long enough to cause core uncovery.

B - WRONG: a second centrifugal pump running in PARALLEL with the existing one will NOT improve HPSI capability. The problem is RCS pressure.

C - WRONG: Maximizing Charging with no letdown flow will NOT be adequate to restore inventory control. The stem states that RVLMS has just dropped to 29% with the 3 charging pumps running. The open PORV equates to the "worse case" size SB-LOCA for MP2, based on the continual loss of inventory being greater than charging pump capacity but lower than core heat removal requirements.

References

EOP 2532, Lesson Plan, R1, C1, pg. 14 of 23:
D. Perform a Controlled Cooldown [EO# - RO-9B]
1. The cooldown should be initiated within 1hr of the event to conserve condensate inventory and comply with the long term cooling analysis. A minimum cooldown rate of 40°F/hr should be applied to ensure that at the 8 - 10hr point of the event boron precipitation conditions can be met. The starting point for the cooldown is the point at which the RCS stabilizes initially after the LOCA. Tc is used for monitoring RCS cooldown. However, if reflux boiling is occuring (occurs with a partially voided hot leg where steam leaves the core region, travels to the steam generator tubes, condenses, and the condensate flows back to the core via the hot leg) CETs should be utilized.

2. In the case of a SBLOCA that is inhibiting Safety Injection, cooldown initiation is extremely important. Delaying the cooldown will prolong the duration of the LOCA with insufficient injection flow and thereby raise the potential for core damage due to voiding and inadequate core heat removal. In this case the operators must take actions to initiate a cooldown at the maximum rate as expeditiously as possible.

Comments and Question Modification History

B -- not plausible.
[Replaced choice "B" - RLC]
{The stem states that RVLMS has just dropped to 29% with the 3 charging pumps running. The open PORV equates to the "worse case" size SB-LOCA for MP2, based on the continual loss of inventory being greater than charging pump capacity but lower than core heat removal requirements. - RLC}

NRC K/A System/E/A

System 008 Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)

Number AK1.02 **RO** 3.1 **SRO** 3.7 **CFR Link** (CFR 41.8 / 41.10 / 45.3)

Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident: Change in leak rate with change in pressure

2 Origin: Mod

☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order?

☑ Selected for Exam ☐ Past NRC Exam?

Question ID: 6054015

NRC K/A Generic

System Number	RO	SRO	CFR Link

2 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 0054015**

Parent ☐ Selected for Exam ☐ Past NRC Exam?

The following conditions exist:|| - A small break LOCA is in progress.| - RCS pressure has stabilized near the shutoff head of the HPSI pumps.| - RVLMS indicates 29% and is dropping.| - Pressurizer level is 0% and stable.||Which one of the following actions will be most effective in ensuring that HPSI flow will become greater than break flow, (i.e. RCS inventory will recover)?|

A Start the third HPSI pump. | ☐

B Commence a cooldown by steaming the steam generators. | ☑

C Initiate Pressurizer Spray. | ☐

D Open the reactor vessel head vents. | ☐

Justification

Starting third HPSI will not significantly increase flow if near shutdoff head; Opening PORVs or head vents may not reduce pressure enough to ensure adequate HPSI flow and also results in further inventory loss; steaming reduces RCS temperature and pressure and is directed in all 2532 evolutions.
(REF EOP-2532 and CEN 152)

References

Comments and Question Modification History

NRC K/A System/E/A

System Number	RO	SRO	CFR Link

NRC K/A Generic

System Number	RO	SRO	CFR Link

3 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Mod ☑ Selected for Exam ☐ Past NRC Exam? **6054020**

The following conditions exist:

- A small break LOCA is in progress.
- All ESAS equipment is operating as designed.
- Moments ago, the Crew commenced steaming both Steam Generators.
- Reactor vessel level has just dropped to 0%.
- Both Cold Leg temperatures are ~ 563 °F and stable.
- Both Hot Leg temperatures are ~ 564 °F and rising slowly.
- RCS Pressure is ~ 1170 psia and steady.

Which one of the following describe what these plant parameters indicate?

A The core is now uncovered with inadequate core cooling from any source. ☐

B The core is covered and being cooled by single phase natural circulation and SI flow. ☐

C The core is covered and being cooled by two-phase natural circulation and SI flow. ☐

D The core is covered and being cooled by reflux boiling and Safety Injection. ☑

Justification

D - CORRECT: The RCS is at saturation, indicating the core is still covered (i.e.; not superheated). Although vessel level indicates 0%, this indication is still above the core.

A - WRONG: Safety injection flow and heater removal from the steam generators by way of reflux boiling is indicative of adequate inventory.

B - WRONG: Single phase natural circulation is NOT possible with 0% vessel level. The Steam Generator tube are NOT in contact with any water in the hot or cold legs.

C - WRONG: Two phase natural circulation is NOT possible with 0% vessel level. The Steam Generator tube are NOT in contact with any water in the hot or cold legs.

References

Steam Tables and Generic Fundamentals

Comments and Question Modification History

B and C add "and SI". Need more technical references.
[added " and SI flow." to "B" & "C" choices - RLC]

NRC K/A System/E/A

System 009 Small Break LOCA

Number EK1.02 **RO** 3.5 **SRO** 4.2 **CFR Link** (CFR 41.8 / 41.10 / 45.3)

Knowledge of the operational implications of the following concepts as they apply to the small break LOCA: Use of steam tables

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

3 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 0054020**
Parent ☐ Selected for Exam ☐ Past NRC Exam?

The following conditions exist:

- A small break LOCA is in progress.
- All ESAS equipment is operating as designed.
- Reactor vessel level has just dropped to 7%.
- Loop 1 cold leg temperature, (T115), is stable at ~ 530 degrees.
- Loop 2 cold leg temperature, (T125), has rapidly dropped from ~ 530 degrees to ~ 320 degrees.
- Both hot leg temperatures ~ 555 degrees and rising slowly.

What do these temperature responses indicate?

A Natural circulation flow has increased. | ☐

B Reflux boiling has resulted in stagnation of the cold legs. | ☑

C The break is in the loop 2 hot leg. | ☐

D The core is being reflooded by HPSI flow. | ☐

Justification

Indicated temperature in loop 2 cold leg drops due to charging flow when the leg stagnates.

References

Comments and Question Modification History

NRC K/A System/E/A

System Number	RO	SRO	CFR Link

NRC K/A Generic

System Number	RO	SRO	CFR Link

4 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Mod ☑ Selected for Exam ☐ Past NRC Exam? **6500057**

Unit 2 was operating at 100% power when a large break LOCA occurred. Given the following events and conditions:

- RSST is unavailable, both EDGs started and loaded on the LNP.
- "A" Service Water Pump will not restart post-trip.
- "B" Service Water (SW) Pump remains secured.
- 24E is aligned to 24D.
- Unit 3 is in the process of checking availability of supplying power to 24E.
- "B" RBCCW Pump is being restored to service by Maintenance (PM's) and is due back within the hour.
- ALL other equipment is operating as designed.

Then, the "C" RBCCW Pump trips on overload. Maintenance is dispatched to determine the cause of the trip.

Which one of the following statements describes how the above equipment malfunctions will affect the mitigating strategy of this event?

A No effect; both trains of CTMT Spray are available to handle the initial energy release and one train of RBCCW and SW remain for long term cooling. ☐

B The initial containment pressure spike will exceed design pressure due to the unavailability of the CAR fans but long term heat removal will be adequate. ☐

C Bus 24E must be energized from Unit 3 or, the "C" RBCCW Pump must be restored by the time SRAS occurs to ensure continued core cooling. ☑

D To ensure continued core cooling, steps must be taken outside the EOPs, including the alignment of Facility 2 Service Water to Facility 1 RBCCW. ☐

Justification

C - CORRECT: The given conditions will result in a loss of all RBCCW. When SRAS occurs, RBCCW is required to provide flow to at least one SDC Heat Exchanger. If Bus 24E is energized from Unit 3, the "B" RBCCW Pump may be started to supply Facility 2. If the "C" RBCCW Pump is started, then Facility 2 is functional again.

A - WRONG: NO RBCCW is in operation for Heat Removal.

B - WRONG: CTMT Spray will prevent CTMT pressure from exceeding the design pressure. Additionally, long term cooling may not be available due to the loss of RBCCW needed for SRAS.

D - WRONG: It would be acceptable to crosstie facilities in order to restore RBCCW. However, since the "A" SW pump was lost following the LNP and the "B" SW pump is aligned to Facility 2 (24E is aligned to 24D), there was no cooling water available for the "A" EDG and it had to be emergency tripped. Therefore, there is no power for any Facility 1 equipment, including the RBCCW pumps.

References

1. ECC-01-C rev 3 page 19
2. ECC01 figure 6 (M2105-11-98)
3. CSS-00-C rev 4 chg 1 pages 6-8, 37
4. FSAR Figure 25203-3001 Main Single Line Diagram 1-10-92
5. RBC-00-C rev 5 pages 24-25
6. EOP 2532, Rev. 24, Step 38

Comments and Question Modification History

D also correct. Need to see reference for impact of loss of bus 24E on "B" RBCCW Pump.
["D" is NOT correct, see reworded "Justification" - RLC]

NRC K/A System/E/A

System 011 Large Break LOCA

Number EA2.03 **RO** 3.7 **SRO** 4.2 **CFR Link** (CFR 43.5 / 45.13)

Ability to determine or interpret the following as they apply to a Large Break LOCA: Consequences of managing LOCA with loss of CCW

4 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Mod ☑ Selected for Exam ☐ Past NRC Exam? **6500057**

NRC K/A Generic

System
Number RO SRO CFR Link

4 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Parent ☐ Selected for Exam ☐ Past NRC Exam? **5000057**

Unit 2 was operating at 100% power when a large break LOCA (design bases) occurred. Given the following events and conditions:

- The SDC HX "A" RBCCW outlet valve (2-RB-13.1A) jammed shut and will not open on SRAS
- The Unit Substation Transformer 24D7-1X secondary windings open due to a fault and bus 22E is deenergized

Which one of the following statements correctly decribes the effect of this failure on the containment spray system if repairs CANNOT be made?

A The initial containment pressure spike will exceed design pressure and long term heat removal will NOT be adequate ☐

B The initial containment pressure spike will exceed design pressure but long term heat removal will be adequate ☐

C The initial containment pressure spike will not exceed design pressure but long term heat removal will NOT be adequate ☐

D The initial containment pressure spike will not exceed design pressure and long term heat removal will be adequate ☑

Justification

The loss of SDC to the A train of containment spray will prevent that train from removing heat during long term sump recirc operations. The initial pressure spike will be mitigated because both trains of CSS will inject from the RWST during the initial pressurization of containment. Long term cooling requires operation of one complete train of ESF equipment which includes 2 CAR fans plus one CSS train. Loss of the 22E emergency bus will cause CAR fans F14A and F14C to lose power. CAR fans F14B and F14D will run from bus 22F.

CHOICE [A] - NO
Both trains of containment spray will inject and maintain the initial pressure spike below the design threshold for containment. Long term heat removal requires only 1 train of containment spray and 2 CAR fans to remove sufficient heat.

CHOICE [B] - NO
Both trains of containment spray will inject and maintain the initial pressure spike below the design threshold for containment. Long term heat removal will be adequate with 2 CAR fans and train B containment spray.

CHOICE [C] - NO
Long term heat removal from containment will be adequate with 2 CAR fans and Train B of CSS.

CHOICE [D] - YES

References

1. ECC-01-C rev 3 page 19
2. ECC01 figure 6 (M2105-11-98)
3. CSS-00-C rev 4 chg 1 pages 6-8, 37
4. FSAR Figure 25203-3001 Main Single Line Diagram 1-10-92
5. RBC-00-C rev 5 pages 24-25

Comments and Question Modification History

NRC K/A System/E/A

System 005 Residual Heat Removal System (RHRS)
Number K3.06 **RO** 3.1* **SRO** 3.2* **CFR Link** (CFR: 41.7 / 45.6)
Knowledge of the effect that a loss or malfunction of the RHRS will have on the following: CSS

NRC K/A Generic

System
Number **RO** **SRO** **CFR Link**

5 | Origin: New | ☑ R | ☐ SRO | ☐ Student Handout? | ☐ Lower Order? | Question ID: 6000001
☑ Selected for Exam | ☐ Past NRC Exam?

The plant is operating normally at 100% power when high vibration alarms are received on the "B" RCP. Based on the vibration level and procedure guidance, the US orders the following:

- Trip the Reactor
- Trip the Main Turbine
- Trip the "B" RCP

"In that order"

Which one of the following describes why the specific ORDER of the directed actions is so important?

A Tripping the Main Turbine before the RCP ensures the Main Turbine does NOT trip on overspeed due to loss of load when the RCP trips. ☐

B Tripping the reactor before the RCP avoids challenging core integrity due to either loss of or low RCS flow with the reactor at 100% power. ☑

C Tripping the Main Turbine between the reactor and the RCP ensures 25B has a chance to transfer to the RSST before the sudden power surge from the RCP loss takes out 25B on overload. ☐

D Tripping the RCP after the reactor prevents formation of major flow oscillations in the core, resulting in uneven changes in nucleate boiling and subsequently unacceptable radial flux distributions. ☐

Justification

B - CORRECT: for the reason stated.

A - WRONG: The Main Turbine will NOT trip on overspeed due to a loss of one RCP.

C - WRONG: The loss of one RCP will Not cause a large enough sudden power surge to cause the bus supply to trip on overload, locking out the 6.9 bus and the available condensate pump. The electrical system can handle the power surge caused by one RCP tripping.

D - WRONG: The loss of one RCP will NOT have a significant impact on core flow or cause unacceptable disturbances in the core.

References

ARP 2590C-054, RCP HIGH VIBRATION, Rev. 000-04
(ARP ref. Gives actual sequence of required actions)

MANUAL Trip before AUTO Trip
OP-2260; RPS ANTICIPATORY MANUAL TRIPS
Approaching RPS Auto Setpoints or Various Turbine trips & efforts to regain control is unsuccessful.

Comments and Question Modification History

B replace/reword. (true regardless of stem) Tripping the reactor first avoids challenging either core integrity due to loss of flow or the low flow trip.
[reword Choice "B" per suggestion]
** Consider rewording "C" to somehthing that actually does not happen. **
{Reworded choice "C" per In-House review - RLC}

NRC K/A System/E/A

System 015 Reactor Coolant Pump Malfunctions

Number AK3.03 **RO** 3.7 **SRO** 4.0 **CFR Link** (CFR 41.5,41.10 / 45.6 / 45.13)

Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow) : Sequence of events for manually tripping reactor and RCP as a result of an RCP malfunction

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

6 | Origin: ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? | Question ID: 6053719
Mod ☑ Selected for Exam ☐ Past NRC Exam?

The plant has been shutdown for refueling for approximately eight (8) days, with Fac. 1 Shutdown Cooling presently in service and all temperatures are stable. The "B" RBCCW header is secured, isolated and about to be drained to do a weld repair on the -25' level of the Aux. Building.

Approximately five (5) minutes after the Control Room is notified that personnel have begun draining the "B" RBCCW header, the crew notices that the "RCS return from SDC" temperature is beginning to rise with NO operator action.

Which of the following describes other indications that could be used to diagnose the probable CAUSE of the SDC temperature rise?

A RBCCW Surge Tank makeup valve opened in automatic; "A" header level 5% and dropping, "B" header level 50% and stable. ☐

B The "A" RBCCW pump amps are beginning to fluctuate and the "A" header flow and pressure are fluctuating and dropping. ☑

C A PEO touring the "A" Safeguards Room reports hearing an unusually high noise level coming from "A" SDC Heat Exchanger. ☐

D 2-SI-657, "SDC HX Flow Control Valve", is slowly opening and 2-SI-306, "SDC Total Flow Control Valve", is slowly closing. ☐

Justification

B - CORRECT: Cavitating RBCCW pump conditions and loss of header flow and pressure are sure indications that the wrong RBCCW header is being drained, resulting in degraded performance of the in-service RBCCW pump and a loss of SDC/RHR heat sink.

A - WRONG: If a RBCCW header had been prepared for draining, the Surge Tank level must be lowered to ~33%, which is the level of the weir that separates the two facilities, not the normal 50% level. This would require the makeup valve be placed in manual/closed as it is preset to maintain 50% level in the surge tank.
VALID DISTRACTOR: because the applicant may expect the RBCCW Surge tank to indicate auto-makeup if the "A" header is accidentally drained, with the Makeup valve maintaining the level normal on the "B" header side based on the presence of the weir.

C - WRONG: This would be indicative of excessive SDC flow or pump noise being reflected through the system to the heat exchanger (largest load).
VALID DISTRACTOR: because the applicant may conclude the abnormal noise is caused by SDC flow automatically adjusting for the loss of heat sink or abnormal RBCCW pump noise being reflected into the largest heat exchanger on the system.

D - WRONG: SDC valve, as presently configured, will not auto-adjust for rising temperatures due to the degrading heat sink.
VALID DISTRACTOR: because the applicant may consider 2-SI-306 auto-response to changing SDC total flow and assume both SDC control valves will respond to changing demands on the SDC system. 2-SI-657 can NOT be operated in automatic temperature control.

References

Generic Fundamentals: indication of a pump losing NPSH (covered in Loss Of SDC/RHR Training Materials and AOP for the SDC Pump).

Comments and Question Modification History

{Modified "A" to put the "B" header level indication at 50%, indicative of the weir holding this header close to the setpoint for auto makeup. Also, corrected valve number in choice "D" from 2-SI-302 to 2-SI-306 - RLC}

NRC K/A System/E/A

System 025 Loss of Residual Heat Removal System (RHRS)

Number AA1.04 **RO** 2.8* **SRO** 2.6 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the Loss of Residual Heat Removal System: Closed cooling water pumps

NRC K/A Generic

6

Origin: Mod

☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order?

☑ Selected for Exam ☐ Past NRC Exam?

Question ID: 6053719

System Number

RO SRO CFR Link

6 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Parent ☐ Selected for Exam ☐ Past NRC Exam? **0053719**

Which one of the following is the reason why RBCCW flow to the SDC heat exchanger (HX) is throttled to a maximum of 4800 gpm?|

A Prevent excessive fluid velocities which could cause damage to the HX by erosion and/or vibration. | ☑

B Minimize possibility of insufficient cooling capability of other essential RBCCW components. | ☐

C Minimize possibility of RBCCW pump reaching runout conditions. | ☐

D Minimize total RBCCW flow in the event of a SIAS. | ☐

Justification

OP 2310, Shutdown Cooling System, PRECAUTION. Design limit for components.

References

Comments and Question Modification History

NRC K/A System/E/A

System Number	RO	SRO	CFR Link

NRC K/A Generic

System Number	RO	SRO	CFR Link

7 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? **Question ID:**
Mod ☑ Selected for Exam ☐ Past NRC Exam? **6500015**

The reactor is shutdown, maintaining Hot Standby conditions.

The following indications for "A" RCP are noted by the PPO:
- Bleedoff temperature - 197 °F and rising
- Lower Seal temperature - 150 °F and stable
- Motor Stator Temperature - 210°F and slowly rising

Given these conditions, identify the correct response related to conditions of "A" RCP.

A Trip pump required because Bleedoff Temperature exceeds the limit ☑

B Trip pump NOT required because no RCP limits are exceeded. ☐

C Trip pump because Lower Seal Temperature exceeds the limit. ☐

D Trip pump because Motor Stator Temperature exceeds the limit. ☐

Justification

A CORRECT: ARP 2590B-070, step 4, requires the pump to be tripped when the annunciator alarms. Also, AOP-2564, Loss Of RBCCW, step 3.3, bullet #6, Page 7 of 46, gives parameters to be monitored, and associated contingency actions required, if a parameter (temperature) is exceeded based on the loss of cooling water.

B - WRONG: Bleedoff temperature exceeded the limit for a pump trip.

C - WRONG: The Lower Seal Temperature did NOT exceed the required trip limit.

D - WRONG: The Stator Temperature did NOT exceed the required trip temperature.

References

1. ARP-2590B-070, RCP A BLEED-OFF TEMP HI, Rev. 000, Alarm setpoint is 180°F. Procedure requires a pump trip at 195°F
2. ARP-2590B-066, "RCP A STR TEMP HI", Rev. 000, Alarm setpoint is 260°F. Procedure requires a pump trip above 260°F.
3. ARP-2590B-079, "RCP A LOWER SEAL TEMP HI", Rev. 000, Alarm setpoint is 170°F. Procedure requiers a pump trip at 170°F rising
4. AOP-2564, Loss Of RBCCW, step 3.3, bullet #6, Page 7 of 46, parameters to be monitored, and associated contingency actions required, on Bleedoff Flow high temp due to RBCCW loss.

Comments and Question Modification History

NRC K/A System/E/A

System 026 Loss of Component Cooling Water (CCW)

Number AA2.04 **RO** 2.5 **SRO** 2.9* **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The normal values and upper limits for the temperatures of the components cooled by CCW

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

7

Origin: ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 5000015**

Parent ☐ Selected for Exam ☐ Past NRC Exam?

The reactor is shutdown, maintaining Hot Standby conditions.

The following are indications for RCP 1A:
- Motor Vibration ==> 0.001 inches peak to peak
- Seal Bleed-Off Flow ==> 0.95 gpm
- Upper Oil Reservoir Level ==> 82.5%
- Motor Stator Temperature ==> 265°F

Given these conditions, identify the correct response related to conditions of RCP 1A.

A Trip pump because motor vibration exceeds limit ☐

B Trip pump not required because nothing exceeds limit ☐

C Trip pump because oil reservoir level exceeds limit ☐

D Trip pump because stator temperature exceeds limit ☑

Justification

CHOICE (A) - NO
WRONG: Motor vibration alarm setpoint is 0.002 to 0.005 inches peak-to-peak. In this range an other groups are contacted for determination as to whether or not pump should be shutdown.
VALID DISTRACTOR: because the applicant may think that vibration is excessive and requiring a pump shutdown.

CHOICE (B) - NO
WRONG: Trip is required due to high stator temperature.
VALID DISTRACTOR: because the applicant may all parameters within specification

CHOICE (C) - NO
WRONG: Oil level is in expected range (75 to 85%). The high level alarm actuates at 87.5%.
VALID DISTRACTOR: because the applicant may think oil level is excessive.

CHOICE (D) - YES
RCP stator temperature is normally 160 to 180 degrees F. The high motor stator temperature alarm actuates at 260 degrees F. Per the alarm response procedure, operators are directed to trip the plant and then the pump above 260 degrees F.

References

1. ARP-2590B-067, "RCP A VIBRATION HI ", Revision 0
2. ARP-2590B-066, "RCP A STR TEMP HI", Revision 0
3. ARP-2590B-082, "RCP A UPR OIL RSVR LEVEL HI", Revision 0
4. CVC-00-C, "Chemical and Volume Control System" Lesson, Revision 8/3 (Pg 14 of 165)

Comments and Question Modification History

NRC K/A System/E/A

System 003 Reactor Coolant Pump System (RCPS)

Number A1.03 **RO** 2.6 **SRO** 2.6 **CFR Link** (CFR: 41.5 / 45.5)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCPS controls including: RCP motor stator winding temperatures

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

8 **Origin:** New ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 6000002**

☑ Selected for Exam ☐ Past NRC Exam?

The plant is at 83% power and stable, awaiting the repair and return of the "B" Condensate Pump. Per OP 2204, the following other plant conditions exist:

- two (2) charging pumps are operating in manual (Mode Switch in LVL-1).
- Pressurizer backup heaters have been manually energized to force sprays.
- PZR pressure has been stabilized at the normal operating pressure.

Then, unknown to the Crew, the in-service Pressurizer (PZR) pressure control transmitter fails AS-IS (output can NOT change).

Which one of the following describes the system response if Tcold were to NOW rise two (2) degrees due to a primary-secondary power mismatch?

A The running backup charging pump would stop. ☐

B PZR pressure and Spray flow would rise slightly. ☑

C The proportional heaters would go to minimum. ☐

D Pressurizer level setpoint would rise to maximum. ☐

Justification

B - CORRECT: PZR pressure would rise only slightly due to an insurge because while forcing PZR spray flow there is no "dead-band" for the PZR pressure controller. As soon as pressure started to rise, the pressure controller would immediately open the spray valves further and stop the rise at some value based on the controllers proportional band.

A - WRONG: Pressurizer level would Not rise enough to cause the backup Charging Pump to stop (3.6% required).

C - WRONG: The Proportional Heaters are already at minimum due to forcing sprays.

D - WRONG: If the controller fails AS-IS, then the level controller will NOT see any change in actual level; therefore, the setpoint (or output) will NOT change.

References

MSS-00-C, Main Steam System (Training Mat.), Rev. 6, Page 47 of 69.
PLPCS, Pressurizer Level and Pressure Control System (Training Mat.), Rev. 3, Page 41 of 61.
ARP 2590B, A-39, Rev. 000-02, PRESSURIZER CH Y LEVEL HI/LO.

Comments and Question Modification History

Need thumbrule for 2°F = 30 psia.
**** Technically incorrect, PZR pressure would NOT rise as stated because while forcing PZR spray flow there is no "dead-band" for the PZR pressure controller. Reword choice "B" to be CORRECT ****
{Reworded Choice/Answer "B" to be correct per the conditions of the stem. Also modified stem so plant conditions are in a list. - RLC}

NRC K/A System/E/A

System 027 Pressurizer Pressure Control System (PZR PCS) Malfunction

Number AK1.02 **RO** 2.8 **SRO** 3.1 **CFR Link** (CFR 41.8 / 41.10 / 45.3)

Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions: Expansion of liquids as temperature increases

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

8 Origin: ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Org ☐ Selected for Exam ☐ Past NRC Exam? **6000002**

The plant is at 83% power and stable, awaiting the repair and return of the "B" Condensate Pump. Per OP 2204, two charging pumps are operating in manual (Mode Switch in LVL-1) and the Pressurizer backup heaters have been manually energized to force sprays.

Then, unknown to the Crew, the in-service Pressurizer pressure control transmitter fails AS-IS (output can NOT change).

Which one of the following describes the system response if Tcold were to NOW rise two (2) degrees due to a primary-secondary power mismatch?

A The running backup charging pump would stop. ☐

B RCS pressure would rise approximately 30 psi. ☑

C The proportional heaters would go to minimum. ☐

D Pressurizer level setpoint would rise to maximum. ☐

Justification

B - CORRECT: As a thumb rule, Pressurizer pressure will rise approximately 15 psia for every 1°F rise in RCS temperature; therefore, a 2°F rise in T cold would result in a 30 psia rise in pressure.

A - WRONG: Pressurizer level would Not rise enough to cause the backup Charging Pump to stop (3.6% required).

C - WRONG: The Proportional Heaters are already at minimum due to forcing sprays.

D - WRONG: If the controller fails AS-IS, then the level controller will NOT see any change in actual level; therefore, the setpoint (or output) will NOT change.

References

MSS-00-C, Main Steam System (Training Mat.), Rev. 6, Page 47 of 69

ARP 2590B, A-39, Rev. 000-02, PRESSURIZER CH Y LEVEL HI/LO

Comments and Question Modification History

Need thumbrule for 2°F = 30 psia.
**** Technically incorrect, PZR pressure would NOT rise as stated because while forcing PZR spray flow there is no "dead-band" for the PZR pressure controller. Reword choice "B" to be CORRECT ****

NRC K/A System/E/A

System 027 Pressurizer Pressure Control System (PZR PCS) Malfunction

Number AK1.02 **RO** 2.8 **SRO** 3.1 **CFR Link** (CFR 41.8 / 41.10 / 45.3)

Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions: Expansion of liquids as temperature increases

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

9 | Origin: New | ☑ R | ☐ SRO | ☐ Student Handout? | ☑ Lower Order? | Question ID: 6000003
☑ Selected for Exam | ☐ Past NRC Exam?

The plant is operating at 100% power when a major earthquake causes a main feedwater pump to trip.

The US has ordered a plant trip on low Steam Generator level, but numerous malfunctions have prevented the crew from deenergizing the CEDS busses by any means available in the control room. All CEAs remain fully withdrawn.

Which of the following describes the required procedure action, or automatic system action, that would now occur to prevent core damage?

A The control operators will manually initiate boron injection immediately to shut down the reactor. ☑

B High RCS pressure will cause Aux. Feedwater to maintain a heat sink for the at-power reactor. ☐

C An automatic main turbine trip on SG level will trigger an alternate means of tripping the reactor. ☐

D The control operators will immediately initiate SIAS to start boron injection and a reactor shutdown. ☐

Justification

A - CORRECT: EOP 2525 requires the operator to manually initiate Emergency Boration on a plant trip where two or more CEAs do NOT fully insert.

B - WRONG: Auxiliary Feed will be automatically actuated by this condition; however, Auxiliary Feedwater flow is NOT adequate to remove heat from the RCS at greater than approximately 12% power. Failure of all CEAs to insert will result in power remaining much greater than 12%.

C - WRONG: A Reactor trip signal will be generated by the Turbine trip, but that signal is processed the same way as the Low SG Level trip and the manual trip, which has been stated in the stem to have failed; therefore, the Reactor will still remain at power.

D - WRONG: Initiation of SIAS will start Boron injection, but this is NOT a procedurally directed action. Initiating SIAS will immediately isolate letdown flow. This will compound the problem because there is not enough room in the PZR steam space at NOP/NOT to inject enough boric acid to adequately shutdown the reactor. Additionally, starting HPSI Pumps will NOT provide any additional Boric Acid flow until RCS pressure is reduced below HPSI shutoff head. This method of Boric Acid injection will NOT shut down the Reactor any faster than the manual initiation of Emergency Boration.

References

EOP-2525; Immediate Actions, Rev. 20, Contingency for ensuring reactor is tripped
EOP-2540A; Functional Revcovery, Reactivity Safety Function guidance for loss.

Comments and Question Modification History

NRC K/A System/E/A

System 029 Anticipated Transient Without Scram (ATWS)

Number EK1.03 **RO** 3.6 **SRO** 3.8 **CFR Link** (CFR 41.8 / 41.10 / 45.3)

Knowledge of the operational implications of the following concepts as they apply to the ATWS: Effects of boron on reactivity

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

9 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Org ☐ Selected for Exam ☐ Past NRC Exam? **6000003**

The plant is operating at 100% power when a major earthquake causes a main feedwater pump to trip.

The US has ordered a plant trip on low Steam Generator level, but numerous malfunctions have prevented the crew from deenergizing the CEDS busses by any means available in the control room.

Which one of the following describes the action credited for preventing core damage?

A The control operators will manually initiate boron injection to shut down the reactor. ☑

B High RCS pressure will cause Aux. Feedwater to maintain a heat sink for the reactor. ☐

C A main turbine trip on SG level will trigger an alternate means of tripping the reactor. ☐

D The control operators will initiate SIAS to start boron injection and a reactor shutdown. ☐

Justification

A - CORRECT: EOP 2525 requires the operator to manually initiate Emergency Boration on a plant trip where two or more CEAs do NOT fully insert.

B - WRONG: Auxiliary Feed will be automatically actuated by this condition; however, Auxiliary Feedwater flow is NOT adequate to remove heat from the RCS at greater than approximately 12% power. Failure of all CEAs to insert will result in power remaining much greater than 12%.

C - WRONG: A Reactor trip signal will be generated by the Turbine trip, but that signal is processed the same way as the Low SG Level trip and the manual trip, which has been stated in the stem to have failed; therefore, the Reactor will still remain at power.

D - WRONG: Initiation of SIAS will start Boron injection, but this is NOT a procedurally directed action. Initiating SIAS will immediately isolate letdown flow. This will compound the problem because there is not enough room in the PZR steam space at NOP/NOT to inject enough boric acid to adequately shutdown the reactor. Additionally, starting HPSI Pumps will NOT provide any additional Boric Acid flow until RCS pressure is reduced below HPSI shutoff head. This method of Boric Acid injection will NOT shut down the Reactor any faster than the manual initiation of Emergency Boration.

References

EOP 2525, Rev. 20, Step 1.c.1 - Determine Status of Reactivity Control; CONTINGENCY ACTIONS

Comments and Question Modification History

NRC K/A System/E/A

System 029 Anticipated Transient Without Scram (ATWS)

Number EK1.03 **RO** 3.6 **SRO** 3.8 **CFR Link** (CFR 41.8 / 41.10 / 45.3)

Knowledge of the operational implications of the following concepts as they apply to the ATWS: Effects of boron on reactivity

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

10 Origin: Mod ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? ☑ Selected for Exam ☐ Past NRC Exam?

Question ID: 6100204

The plant has tripped due to a loss of the electrical grid. On the trip, both facilities of Vital AC were repowered by their respective EDG. All other plant systems and components responded as designed.

Five (5) minutes into the performance of EOP-2525, a Steam Generator Tube Rupture (SGTR) occurred on #1 SG. The Crew has now completed EOP-2525 and the US is about to direct the Board Operators to commence a cooldown per EOP-2534, SGTR.

Then, VA-10 deenergizes when it's main breaker fails open. PEOs are now being sent out in the plant to perform various tasks due to the loss of VA-10.

Which of the following local PEO tasks would require direct HP coverage due to ALARA concerns from the SGTR?

A Verification of the continued operation of the EDGs as designed. ☐

B Local-Manual operation of the "A" ADV during the plant cooldown. ☑

C Local-Manual operation of the "A" Aux. Feedwater Reg. Valve. ☐

D Swapping the "B" charging pump from facility 1 to facility 2 power. ☐

Justification

B - CORRECT: Local-Manual operation of the "A" ADV will result in above normal exposure to the PEO due to the radioactive steam of the #1 SGTR.

A - WRONG: Although checking of the EDGs requires passing through the Aux. Building control point, the SGTR has no effect on background radiation in any of the areas or pathways utilized for check both EDGs.

C - WRONG: It is unlikely any contaminated steam made it into the Turbine Building with the plant tripping on a LOOP five minutes before the SGTR occurred. The SPO should have closed MSIVs before the SGTR due to the LNP. However, even if some amount of contaminated steam passed into the Turbine Building, the location of the #1 AFRV is not near any component that would now be an increased source of exposure.

D - WRONG: The "B" charging pump Kirk-Key interlock is located in the same area as the charging pumps, however, a SGTR does not effect the rad. Levels in this area like a LOCA or any type of event involving fuel damage would.

References

OP-2316A (Main Steam System), Rev. 032-02, Page 10 of 106
E34-01-C (EOP-2534, SGTR Training Material), Rev. 2, Ch. 2; Concerns for SGTR

Comments and Question Modification History

K./A mismatch. Question could be answererd without initial sentence. Align w/ SGTR
[Modified question - RLC]

NRC K/A System/E/A

System 038 Steam Generator Tube Rupture (SGTR)

Number GA **RO** **SRO** **CFR Link**

SEE GENERIC K/A

NRC K/A Generic

System 2.3 Radiation Control

Number 2.3.2 **RO** 2.5 **SRO** 2.9 **CFR Link** (CFR: 41.12 / 43.4. 45.9 / 45.10)

Knowledge of facility ALARA program.

10 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Parent ☐ Selected for Exam ☐ Past NRC Exam? **1000204**

The plant has been shut down due to a leak in the CVCS Regenerative Heat Exchanger. You have been directed to open and red tag the vents and drains associated with the heat exchanger. The following conditions exist:

- The area around the CVCS Regenerative Heat Exchanger has been posted a 'Locked High Radiation Area'.
- The area is at the MINIMUM required radiation level for the posting.
- Your present annual exposure is 500 mRem.
- All the valves you have been assigned to operate are inside the posted area.

What would be your maximum calculated stay time in this area in order to avoid exceeding the Millstone administrative limit?

A 4.5 hours ☐

B 2.5 hours ☐

C 1 hour ☐

D 30 minutes ☑

Justification

D is correct. The Millstone administrative limit, based on ALARA considerations, is 1,000 mRem/yr; therefore, your maximum allowed exposure is 500 mRem. The minimum radiation level for a 'Locked High Radiation Area' is 1,000 mRem/hr. The maximum allowed time is 30 minutes. (500 mRem divided by 1,000 mRem/hr = 0.5 hr. 0.5 hr x 60 minutes/hr = 30 minutes)

A is incorrect. If the administrative limit is assumed to be 5,000 mRem/yr (4,500 mRem left for the year), then the maximum stay time is 4.5 hrs.

B is incorrect. If the assumed Millstone administrative limit is 3,000 mRem/yr (2,500 mRem left for the year), then the maximum stay time is 2.5 hrs.

C is incorrect. If the lock high radiation area minimum value is assumed to be 500 mRem, then the maximum stay time is 1 hr.

References

Comments and Question Modification History

NRC K/A System/E/A

System 2.3 Radiation Control

Number G **RO** **SRO** **CFR Link**

SEE GENERIC K/A

NRC K/A Generic

System 2.3 Radiation Control

Number 2.3.2 **RO** 2.5 **SRO** 2.9 **CFR Link** (CFR: 41.12 / 43.4. 45.9 / 45.10)

Knowledge of facility ALARA program.

11

Origin: New — ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? ☑ Selected for Exam ☐ Past NRC Exam?

Question ID: 6000004

The plant is at 100% power when it trips due to a loss of Battery Bus 201B. On the trip, a large rupture occurs on the "B" Main Steam Header inside Containment.

Which of the following describes the plant system design, if any, and a response that will automatically mitigate this event?

A Aux. Feed Water requires power to actuate, so the loss of power to Facility 2 AFW will automatically prevent AFW from feeding the ESD. ☐

B The Turbine Battery will power VA-20 to ensure the #2 Main Feed Reg. Valve closes on a MSI signal, preventing main feed flow to the ESD. ☑

C All ESAS cabinets have dual power supplies to ensure both ESAS facilities are available to automatically actuate all necessary components. ☐

D This condition is beyond the design basis of the plant and there are NO system or plant components that will automatically mitigate the event. ☐

Justification

B - CORRECT The Turbine Battery is the back up power to VA-20 through Inverter 6. Maintaining VA-20 energized will allow the #2 Main Feed regulating Valve to automatically close on an MSI.

A - WRONG: The loss of Facility 2 DC power will cause the #2 Aux Feed Regulating Valve to fail open, and admit the maximum feed flow to the #2 Steam Generator.

C - WRONG: This statement is NOT true; however, The sensor cabinets do have dual power supplies; however, the actuation cabinets have one power supply. Additionally, this has nothing to do with mitigation of an ESD.

D - WRONG: This condition is NOT beyond design basis.

References

CLP 125VDC/120VAC (TDC-00-C) Rev. 2, Chg. 1; Page 9 of 81

Comments and Question Modification History

NRC K/A System/E/A

System E05 Excess Steam Demand

Number EK2.1 **RO** 3.3 **SRO** 3.6 **CFR Link** (CFR: 41.7 / 45.7)

Knowledge of the interrelations between the Excess Steam Demand and components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

12

Origin: ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 1000011**

Bank ☑ Selected for Exam ☐ Past NRC Exam?

The plant is at 100% power with the "B" Auxiliary Feedwater (AFW) pump out of service.

Then, the following plant events occur:

- The main feeder breaker on Vital DC distribution panel trips open on a bus fault, de-energizing DV-10.
- The plant trips on high Pressurizer pressure due to the closure of both MSIVs.
- On the plant trip, the "A" Main Steam Header ruptures in CTMT.

Which of the following actions are required to establish at least one Steam Generator as a controllable RCS heat sink?

A Ensure the "A" AFW pump is running, close 2-FW-44, close 2-FW-44, take manual control of #2 AFW Reg. Valve and feed only #2 SG. ☐

B Ensure the "A" AFW pump is running and feed only #2 SG with it's AFW Reg. Valve in Local-Manual in the Turbine Building. ☐

C Start the Turbine Driven AFW pump, place the #1 AFW Reg. Valve in override and feed only #2 SG using the #2 AFW Reg. Valve. ☐

D Start the Turbine Driven AFW pump, close 2-FW-44, take manual control of the #2 AFW Reg. Valve and feed only the #2 SG. ☑

Justification

D - CORRECT; The TDAFP is the only AFW pump available once 2-FW-44 is closed. FW-44 isolates the discharge header to #1 SG AFW Reg. Valve, which cannot be controlled from the control room and has failed open due to the loss of DV-10.

A - WRONG; 150 gpm per SG is within the pump's capacity, but DV-10 supplies control power to the pump breaker.

B - WRONG; 150 gpm per SG is within the pump's capacity, but DV-10 supplies control power to the pump breaker, also this addresses the wrong side of FW-44.

C - WRONG; Only in the fact that TDAFP is on the wrong side of FW-44

References

AFW-00-C, Rev. 5, Ch. 3, Pages 11, 18 & 19

Comments and Question Modification History

*** Make Choice "C" more wrong.***
{Reworded Choice "C" to remove it as a viable option - RLC}

NRC K/A System/E/A

System 054 Loss of Main Feedwater (MFW)

Number AA1.02 **RO** 4.4 **SRO** 4.4 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the Loss of Main Feedwater (MFW): Manual startup of electric and steam-driven AFW pumps

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

13 Origin: Mod ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 6056724**

☑ Selected for Exam ☐ Past NRC Exam?

The plant has tripped and suffered a total loss of ALL Vital AC. Vital AC is expected to be lost for at least three hours. Based on this expectation, the crew is working through the required actions of EOP-2530.

Which one of the following describes a required action, and the reason for the action's time restriction, per EOP 2530, Station Blackout?

A If Vital AC is not restored within 60 minutes, open PPC UPS breakers, preventing loss of PPC due to battery backup power degradation. ☐

B Within 60 minutes of commencing a cooldown, and every 50 °F thereafter, verify RCS boron concentration meets shutdown margin. ☐

C If Vital AC is not restored within 30 minutes, align supplemental cooling to all vital areas due to loss of ventilation and ventilation cooling. ☑

D If Vital AC is not restored within 30 minutes, ensure backup air is aligned to applicable valves to preclude loss of valve position control. ☐

Justification

C - CORRECT: The requirement to align supplemental cooling is to limit the heat loading in the Vital DC Switchgear Rooms. This requirement is one of the most important time-critical operator actions in 2530 as it could result in the loss of ALL remaining Safety Channels of instrumentation.

A - WRONG: When the PPC is lost due to slow voltage degradation, the first indication is "BAD" or unjustified indications displayed by parameters monitored by the PPC. The danger here is operators may respond to these indications, not knowing the PPC is slowly going "brain dead". Also, the PPC does NOT "die" well when it is lost due to a slow power supply degradation. It will take MUCH longer to restore when lost in this fashion.

B - WRONG: There is a requirement in 2530 to verify RCS boron concentration meets shutdown margin every 50 °F on a cooldown, but the concentration MUST be verified BEFORE the cooldown is started and for the NEXT 50 °F temperature.

D - WRONG: The backup air available to certain valves is designed to assist in mitigating a plant accident (i.e.; LOOF) combined with a loss of Instrument Air. The Station Blackout event assumes no other plant events require mitigation, except for those directly attributed to the lack of Vital AC.

References

CLP Station Blackout (E30-01-C) Rev. 3, Page 15 of 22

Comments and Question Modification History

Need reference regarding preventing PPC degradation.
[Changed "times" shown on choices so "C" would be correct and "A" would be wrong, per EOP-2530 Training Material. Also, enhanced the "Justifications" for both "A" and "C". - RLC]

NRC K/A System/E/A

System 055 Loss of Offsite and Onsite Power (Station Blackout)

Number EK3.01 **RO** 2.7 **SRO** 3.4 **CFR Link** (CFR 41.5 / 41.10 / 45.6 / 45.13)

Knowledge of the reasons for the following responses as the apply to the Station Blackout: Length of time for which battery capacity is designed

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

13 | Origin: ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? | Question ID:
Parent ☐ Selected for Exam ☐ Past NRC Exam? | 0056724

EOP 2530, Station Blackout, requires action to be taken within 60 minutes for which of the following conditions/functions?

A DC power consumption. | ☑

B RCS inventory control. | ☐

C Loss of ventilation. | ☐

D RCS heat removal. | ☐

Justification

EOP 2530 Caution #2 prior to Step 2.22

References

Comments and Question Modification History

NRC K/A System/E/A

System Number	RO	SRO	CFR Link

NRC K/A Generic

System Number	RO	SRO	CFR Link

14 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
New ☑ Selected for Exam ☐ Past NRC Exam? **6000005**

The plant is at 55% power in preparation for restoring the "A" Main Feedwater Pump to service. The following additional plant conditions exist:

- "C" charging pump is operating as the "running" pump.
- "A" charging pump is running in manual (Mode Switch in LVL-1).
- "B" charging Pump is aligned to Facility Two.
- All Pressurizer backup heaters have been manually energized to force Pressurizer sprays.
- Channel "X" PZR Level and Pressure Control are in service.
- All plant parameters are stable.

Then, the plant trips when Vital DC bus 201B deenergizes due to a faulty supply breaker. The Crew is performing steps in EOP-2525, but has NOT taken any actions OUTSIDE of the Control Room.

Which one of the following describes an expected plant status, and why, as a result of the loss of power and subsequent plant trip?

A Letdown will be isolated (auto) and at least two charging pumps will be running, causing a continuous rise in Pressurizer level and pressure. ☐

B VR-21 will also be deenergized preventing the operation of any charging pumps. Pressurizer level will be dropping continuously due to this loss. ☐

C Ten minutes post-trip, RCS pressure will be slowly rising due to only one bank of proportional heaters and one charging pump being available. ☑

D VA-20 will also deenergize causing Ch. "Y" Pressurizer level to fail low and trip all Pressurizer heaters. This will cause RCS pressure to drop. ☐

Justification

C - CORRECT: The loss of 201B causes a complete loss of Fac. 2 AC power, including 25B, 24B, 24D, 22F and VR-21. When VR-21 deenergizes, the High PZR Pressure interlock for the backup heaters fails in the "triggered" mode, tripping ALL backup heater breakers and preventing them from being used. Even though two groups of backup heaters have energized busses per the given scenario, they are still deenergized by a "failure" in there control circuit and can't be energized as long as VR-21 is dead. With only one Facility of Vital AC available (due to the described loss of DC), this leaves only one bank/group of PZR proportional heaters and available to maintain pressure. Ten minutes post-trip, the PZR will still be "recovering" with only one group of proportional heaters and one charging pump.

A - WRONG: Letdown should not isolate automatically because VR-11 was NOT deenergized and "A" charging pump never stopped (it never lost power and its control circuit also remained energized. However, only one charging pump is available until "B" charging pump power supply is LOCALLY aligned to Fac. One.

B - WRONG: If VR-11 were deenergized all available (powered) charging pumps would be operating due to a power loss to part of their control circuit. If Channel "Y" LEVEL control were in service, the loss of power would deenergize it requiring manual control of the available charging pump.

D - WRONG: VA-20 will swap to its alternate power source, INV-5, which is powered by the Turbine Battery and unaffected. This power supply swap is a "make-before-break" and, therefore, there will NOT be a momentary loss of power to Ch. "Y" PZR level transmitter. This will prevent the heaters from tripping on a "failed" LO-LO LEVEL trigger.

References

AOP-2504B, Rev. 003-04, Loss of VR-21; Page 4 of 52 - "Discussion" section.

Comments and Question Modification History

Show Gill how pressure could be rising under this condition.
[I will verify this condition with co-workers, however, I believe 10 minutes into a trip, RCS temperature has long since stabilized (i.e.; stopped dropping). Also, the MSIVs go shut on the loss of DC, so the RCS cooldown on the trip is going to be a lot slower and not go drop as far due to the higher setpoint of the ADVs versus the Condenser Dumps. These conditions should allow any heater or inventory input to cause PZR leve, 10 minutes into a trip, to begin recovery .. I think - RLC]

NRC K/A System/E/A

14 Origin: New ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 6000005**

☑ Selected for Exam ☐ Past NRC Exam?

System 056 Loss of Offsite Power

Number AA2.73 **RO** 3.5 **SRO** 3.6 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Loss of Offsite Power: PZR heater on/off

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

15 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Mod ☑ Selected for Exam ☐ Past NRC Exam? **6055354**

With the plant in Mode 5 during a refueling outage with the following conditions:

VA-10 has been placed on it's ALTERNATE source for electrical PM's on Inverter 1 (INV-1). The PMs have been completed and Inverter 1 and its Static Switch are presently being restored to operation with the following conditions existing:

- The "AUTO/MAN" switch inside the Inverter 1 (INV-1) cabinet is in the MANUAL position.
- The "SYNC" switch on the Static Switch is in the ON position.
- The Inverter is energized with all input and output breakers closed.
- Voltage of the two sources are approximately equal.
- Both sources in synch.

Then, INV-5 trips off line due to an internal fault.

Which one of the following describes the status of, or the required actions for, VA-10?

A When INV-5 deenergized, static switch #1 "auto" transferred VA-10 to its normal power source. ☐

B INV-1 must be completely deenergized (shutdown) and restarted in order to reenergize VA-10. ☐

C The "SYNC" switch on #1 static switch must be placed in the "OFF" position to recover VA-10. ☐

D The "AUTO/MAN" switch inside INV-1 must be placed in the "AUTO" position to recover VA-10. ☑

Justification

D - CORRECT: The nomenclature of the "AUTO/MAN" switch inside INV-1 is similar to the "AUTO/MAN" switch OUTSIDE the INV-1 cabinet on the Static Switch. However, their functions are quite different. With the INV-1 switch specified in the stem in the applicable position, the Static Switch will NOT transfer to the Alternate power supply and VA-10 will be deenergized.

A - WRONG: The transfer is blocked from happening until the above mentioned switch is repositioned.

B - WRONG: Shutting down the inverter completely is NOT required, but if done correctly, would correct the switch misalignment and reenergize VA-10.

C - WRONG: Placing the "SYNC" switch to ON is one of the actions that triggered the actual event that occurred at MP2.

NOTE: This question is similar to an actual scenario that occurred at Millstone Unit 2 and resulted in a loss of RHR and an RCS temperature rise of >10°F (Classifiable event).

References

CLP 125VDC/120VAC (LVD-00-C) Rev. 5, Chg. 1; Page 26 of 81

Comments and Question Modification History

Add to stem: Not complete per procedure.
Add the following to list of "given" info in the stem.
Voltage of the two sources match to ~25%
Both sources in synch.
[Added above suggestions to question stem, but stated "voltages are approximately equal" vs. "within ~25%" for clarity - RLC]

NRC K/A System/E/A

System 057 Loss of Vital AC Electrical Instrument Bus

Number AA1.01 **RO** 3.7* **SRO** 3.7 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the Loss of Vital AC Instrument Bus: Manual inverter swapping

NRC K/A Generic

15 **Origin: Mod** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 6055354**

☑ Selected for Exam ☐ Past NRC Exam?

System Number	RO	SRO	CFR Link

15

Origin: ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 0055354**

Parent ☐ Selected for Exam ☐ Past NRC Exam?

With the plant at 100% normal line-up, the following conditions exist:||o VA-10 is on it's ALTERNATE source.|o Inverter 1 is energized with no alarms|o "AUTO/MAN" switch inside the Inverter 1 cabinet is in MANUAL.||What condition will cause VA-10 to be supplied from it's NORMAL source?||

A Alternate Source deenergizes. | ☐

B "AUTO/MAN" switch is placed in the "AUTO" position. | ☑

C Inverter-1 receives an "Out of Sync" condition. | ☐

D Alternate Source experiences a low voltage condition. | ☐

Justification

$$With the "AUTO/MAN" switch in the "AUTO" position, the static switch will automatically transfer from the Alternate Source to the Normal Source if the Normal Source is in Sync and has no alarms.$$$$Question also correlates to Objectives 226 and 980$$$$$$

References

Comments and Question Modification History

NRC K/A System/E/A

System Number	RO	SRO	CFR Link

NRC K/A Generic

System Number	RO	SRO	CFR Link

16 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Mod ☑ Selected for Exam ☐ Past NRC Exam? **6100078**

The plant has tripped from 100% power with the following conditions:

- ALL electrical components and busses were aligned for normal operation.
- Loss of off-site power (cause of plant trip).
- On the trip, the "B" Battery Bus tie to 201B trips open due to a breaker failure.
- All other plant components and systems are operating as designed.

Which of the following describes indications that would result from the above loss of electrical busses?

A 201A indicates: 134 Battery Volts, -5 Battery Amps and +15 Bus Amps. ☐
201B indicates: 0 Battery Volts, 0 Battery Amps and 0 Bus Amps.

B 201A indicates: 126 Battery Volts, +20 Battery Amps and +15 Bus Amps. ☐
201B indicates: 126 Battery Volts, +20 Battery Amps and +20 Bus Amps.

C 201A indicates: 134 Battery Volts, -5 Battery Amps and +15 Bus Amps. ☑
201B indicates: 134 Battery Volts, 0 Battery Amps and 0 Bus Amps.

D 201A indicates: 134 Battery Volts, -5 Battery Amps and +15 Bus Amps. ☐
201B indicates: 126 Battery Volts, -5 Battery Amps and +15 Bus Amps.

Justification

C - CORRECT:
201A indicates: 134volts indicates a battery on float charge with an indicated 5 amp float and 15 amps worth of load on the bus (normal state);
201B indicates: 134volts indicates a battery on float charge with an indicated 0 amp current flow (battery is isolated) and 0 amps of load on the bus (Bus is isolated from any power source).

A - WRONG:
201A indicates: 134volts indicates a battery on float charge with an indicated 5 amp float and 15 amps worth of load on the bus (normal state);
201B indicates: 0 battery volts would indicate the battery bus "fuses" have blown in the disconnect.

B - WRONG: 126 volts is the low voltage alarm setpoint and would indicate a battery in being drained due to a lack of an energized battery charger, NOT disconnected from the load bus. This would be correct if it was believed the battery chargers have been lost due to the breaker failure or LOOP/LNP.

D - WRONG: This has the expected indication for 201A, but indicates 201B is still connected and being discharged at a normal rate.

References

CLP 125VDC/120VAC (LVD-00-C) Rev. 5, Chg. 1; Page 14 of 81

Comments and Question Modification History

**** Remove reference to "B" Battery Charger as it adds only confusion to the question. ****
{Removed SPECIFIC reference to any battery charger to eliminate confusion - RLC}

NRC K/A System/E/A

System 058 Loss of DC Power

Number AK1.01 **RO** 2.8 **SRO** 3.1* **CFR Link** (CFR 41.8 / 41.10 / 45.3)

Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

16 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Org ☐ Selected for Exam ☐ Past NRC Exam? **6100078**

The plant has tripped from 100% power with the following conditions:

- Loss of off-site power (cause of plant trip).
- On the trip, the "B" Battery Bus tie to 201B trips open due to a breaker failure.
- The "B" Battery Charger is aligned for operation.
- All other plant components and systems are operating as designed.

Which of the following describes indications that would result from the above loss of electrical busses?

A 201A indicates: 134 Battery Volts, -5 Battery Amps and +15 Bus Amps. ☐
201B indicates: 0 Battery Volts, 0 Battery Amps and 0 Bus Amps.

B 201A indicates: 126 Battery Volts, +20 Battery Amps and +15 Bus Amps. ☐
201B indicates: 126 Battery Volts, +20 Battery Amps and +20 Bus Amps.

C 201A indicates: 134 Battery Volts, -5 Battery Amps and +15 Bus Amps. ☑
201B indicates: 134 Battery Volts, 0 Battery Amps and 0 Bus Amps.

D 201A indicates: 134 Battery Volts, -5 Battery Amps and +15 Bus Amps. ☐
201B indicates: 126 Battery Volts, -5 Battery Amps and +15 Bus Amps.

Justification

C - CORRECT:
201A indicates: 134volts indicates a battery on float charge with an indicated 5 amp float and 15 amps worth of load on the bus (normal state);
201B indicates: 134volts indicates a battery on float charge with an indicated 0 amp current flow (battery is isolated) and 0 amps of load on the bus (Bus is isolated from any power source).

A - WRONG:
201A indicates: 134volts indicates a battery on float charge with an indicated 5 amp float and 15 amps worth of load on the bus (normal state);
201B indicates: 0 battery volts would indicate the battery bus "fuses" have blown in the disconnect.

B - WRONG: 126 volts is the low voltage alarm setpoint and would indicate a battery in being drained due to a lack of an energized battery charger, NOT disconnected from the load bus. This would be correct if it was believed the battery chargers have been lost due to the breaker failure or LOOP/LNP.

D - WRONG: This has the expected indication for 201A, but indicates 201B is still connected and being discharged at a normal rate.

References

CLP 125VDC/120VAC (LVD-00-C) Rev. 5, Chg. 1; Page 14 of 81

Comments and Question Modification History

Replace "expected" in stem.
[removed "expected" and "the" before "indications" - RLC]
{Modified stem to remove unnecessary reference to "C" Battery Charger. - RLC}

NRC K/A System/E/A

System 058 Loss of DC Power

Number AK1.01 **RO** 2.8 **SRO** 3.1* **CFR Link** (CFR 41.8 / 41.10 / 45.3)

Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

16 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 1000078**

Parent ☐ Selected for Exam ☐ Past NRC Exam?

The US is in EOP 2540B directing the SPO to verify that at least one vital DC bus is available and powered from its associated battery charger.

The SPO refers to the 3 meters associated with bus 201A, "Battery Volts", "Battery Amps", and "Bus Amps".

Which of the following sets of readings indicates that bus 201A is supplying Facility 1 DC loads and is powered from its associated charger?

A "Battery Volts" 126 "Battery Amps" +20 "Bus Amps" +20 ☐

B "Battery Volts" 132 "Battery Amps" +20 "Bus Amps" +20 ☐

C "Battery Volts" 134 "Battery Amps" -5 "Bus Amps" 0 ☐

D "Battery Volts" 134 "Battery Amps" -5 "Bus Amps" +15 ☑

Justification

D: CORRECT: 134volts indicates a battery on float charge with an indicated 5 amp float and 15 amps worth of load on the bus;

A: WRONG: 126 volts is the low voltage alarm setpoint and both ammeters at +20 indicates a 20 amp drain on the battery;

B: WRONG: 132 volts is a fully charged battery, but without any float charge, both ammeters at +20 indicates a 20 amp drain on the battery;

C: WRONG: 134volts indicates a battery on float charge with an indicated 5 amp float, but there is no load indicated on the DC bus indicating it is stripped or the tie breaker is open

References

Comments and Question Modification History

NRC K/A System/E/A

System 063 DC Electrical Distribution System

Number A3.01 **RO** 2.7 **SRO** 3.1 **CFR Link** (CFR: 41.7 / 45.5)

Ability to monitor automatic operation of the DC electrical system, including: Meters, annunciators, dials, recorders, and indicating lights

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

17 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Mod ☑ Selected for Exam ☐ Past NRC Exam? **6067775**

The plant is operating in MODE 4 shutting down for refueling. The following plant conditions exist:

* "A" & "B" RCPs are operating.
* SDC is presently operating in "Warm-up Recirc" with "SDC to RCS temperature, T351Y" at 125°F and rising.
* RCS pressure is presently 255 psia and stable..

Then, the following control board indications are received:

* "A" RBCCW Header temperature is 105°F and rising.
* RBCCW Surge Tank level is 55% and rising.
* "A" RBCCW Header flow has not changed since entry into MODE 3.
* RBCCW Radiation Monitor indication is stable (NOT in alarm).

Which of the following conditions would result in the above parameters?

A Heat input from "A" Shutdown Cooling (SDC) Heat Exchanger ☐

B A leak from the Spent Fuel Pool (SFP) Cooling Heat Exchanger ☐

C A leak from the "A" Reactor Coolant Pump (RCP) Seal Cooler ☐

D Mussel clogging of the Facility One (1) RBCCW heat exchanger. ☑

Justification

D: CORRECT: Mussel clogging would cause a loss of Service Water flow through the HX (loss of heat sink). In the closed RBCCW System the resulting temperature rise would the surge tank level to go up.

A: WRONG: This would cause the above conditions if the SDC HX was aligned for RBCCW flow. During "Warm-up Recirc" RBCCW is not aligned.

B: WRONG: The SFP cooling system pressure is too low to leak into RBCCW system.

C: WRONG: A leak from a RCP seal cooler would result in RBCCW radiation monitor rise/alarm.

References

ARP-2590E-048, Rev. 000, "RBCCW HX TEMP HI", Step 6; references high temp. caused by HX fouling.

Comments and Question Modification History

NRC K/A System/E/A

System 062 Loss of Nuclear Service Water

Number AA1.05 **RO** 3.1 **SRO** 3.1 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the Loss of Nuclear Service Water (SWS): The CCWS surge tank, including level control and level alarms, and radiation alarm

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

17 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 0067775**
Parent ☐ Selected for Exam ☐ Past NRC Exam?

The following plant conditions exist:||- The plant is operating in MODE 5 for "B" RCP seal replacement.|- Vessel level is at the centerline of the hot leg.|- Nozzle dams are NOT installed.|- A Reactor Building Component Cooling Water (RBCCW) system HIGH radiation alarm has been received on RC-14.|- Chemistry samples indicate the "B" RBCCW header has radioactive in-leakage.||Which of the following is a possible source of this in-leakage?||

A "B" Shutdown Cooling (SDC) Heat Exchanger | ☑

B Spent Fuel Pool (SFP) Cooling Heat Exchanger | ☐

C "D" Reactor Coolant Pump (RCP) Seal Cooler | ☐

D Primary Drain Tank (PDT) and Quench Tank (QT) Cooler | ☐

Justification

The "B" SDC heat exchanger is the only listed RBCCW load on the "B" Header that is at a higher pressure than RBCCW.
[Copied from Item No '5034' on 8/23/96]

References

Comments and Question Modification History

NRC K/A System/E/A

System Number	RO	SRO	CFR Link

NRC K/A Generic

System Number	RO	SRO	CFR Link

18 | Origin: New | ☑ R | ☐ SRO | ☐ Student Handout? | ☑ Lower Order? | Question ID: 6000009

☑ Selected for Exam | ☐ Past NRC Exam?

The plant has tripped from 100% power due to an air leak in the Turbine Building. The crew has transitioned to EOP-2526 and are referencing AOP-2563, Loss Of Instrument Air, for guidance on system restoration. All plant conditions are stable.

Analysis of the leak location using the P&IDs reveals that the Unit 3 air system cross tie could supply the temporarily isolated systems and allow for return of remote operation from the Control Room. However, the Loss of Instrument Air AOP does NOT provide steps for restoration of air once the leak is isolated.

Which of the following describes an administrative requirement that must be met in order to allow the restoration of the Instrument Air System.

A The "on-watch" Unit 2 SM or US must declare entry into 10-CFR-50.54X in order to deviate from the applicable AOP. ☐

B Only Technical Support or Plant Management can authorize this change as it applies to an AOP and not an NOP. ☐

C Senior Licensed personnel from Unit 2 and Unit 3 must approve the procedure change to the Loss of Instrument Air AOP. ☐

D An "on-watch" Unit 2 Senior License must review/approve the procedure change to the Loss of Instrument Air AOP. ☑

Justification

D - CORRECT: This is considered a "Temporary Procedure Change", therefore all aspects of the change require approval by the "on-watch" SM or US in the applicable unit.

A - WRONG: 50.54X can only be implemented if emergency conditions exist, and in order to protect the general public.

B - WRONG: These requirements are only applicable if restoring the plant in a post accident condition.

C - WRONG: SRO review/approval from both units not applicable even though cross-tie affects both units.

References

Minor Procedure Revision; MP-05-MMM, Page 22

Comments and Question Modification History

Is this fair for a RO? Verify learning objective for RO.

Check that "B" is not correct. WC-10 states that rubber hoses are not controlled by other procedures.
[Changed "procedure" to "P&ID" and will verify O.K. for RO with "SM" qualified Instructor - RLC]
{Reworded question and distractors to solicite knowledge of "Procedure Change" versus "Bypass Jumper" - RLC}

NRC K/A System/E/A

System 065 Loss of Instrument Air

Number GA **RO** **SRO** **CFR Link**

SEE GENERIC K/A

NRC K/A Generic

System 2.2 Equipment Control

Number 2.2.11 **RO** 2.5 **SRO** 3.4* **CFR Link** (CFR: 41.10 / 43.3 / 45.13)

Knowledge of the process for controlling temporary changes.

19 | Origin: ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? | **Question ID: 6075164**
Mod ☑ Selected for Exam ☐ Past NRC Exam?

The plant is in normal operation at 100% power, when the Ch."Y" Pressurizer (PZR) Level Controller, LIC-110Y, began to operate erratically.
The PPO has selected Ch. "X" as the controlling channel of PZR level per the applicable ARP and the PZR is now stable at NORMAL, 100% power parameters.
I&C is now working in the applicable Spec-200 cabinet, troubleshooting the LIC-110Y circuit and comparing it's output values to that of LIC-110X circuit.

Then, while I&C is troubleshooting PZR Level Control, the PPO notices the following plant indications:

- PZR Level Ch. "X" PPC indications is slowly rising.
- PZR Level Ch. "Y" PPC indications is slowly lowering.
- PZR Level Controller LIC-110X "red" needles (vertical scale) is slowly rising.
- PZR Level Controller LIC-110Y "red" needles (vertical scale) is slowly lowering.
- PZR Level Controller LIC-110X "black" needle (horizontal scale) is rising.
- PZR Level Controller LIC-110Y "black" needle (horizontal scale) is stable (NOT changing).
- Letdown flow is slowly rising.
- ALL Safety and Control channel indications of PZR Pressure are slowly lowering.
- RCS temperature is stable at 100% power values.

Which of the following describes the malfunction and some of the required actions to be taken, per the applicable ARP, for PZR Level Control?

A Ch. "X" PZR Level TRANSMITTER is failing; the bias output on the letdown flow controller must be lowered to stabilize Pressurizer parameters. ☐

B Ch. "X" PZR Level TRANSMITTER is failing; LIC-110X must be placed in MANUAL and the output lowered to stabilize Pressurizer parameters. ☑

C Ch. "X" PZR Level CONTROLLER is failing, swap level control to LIC-110Y, in AUTOMATIC, and ensure PZR parameters stabilize at normal. ☐

D Ch. "X" PZR Level CONTROLLER is failing, swap level control to LIC-110Y, in MANUAL and lower the output to stabilize Pressurizer parameters. ☐

Justification

B - CORRECT: Pressurizer level is lowering because the Ch. "X" PZR Level TRANSMITTER is failing high. Therefore, LIC-110X must be placed in MANUAL and the output lowered to stabilize Pressurizer parameters.

A - WRONG: Lowering the bias will not restore PZR. Level as the Chg. Pumps are controlled from the level controller.

C - WRONG: The transmitter is failing not the controller. Also, LIC-110Y is not responding to the rising input and therefore must be considered inop due to I&C troubleshooting.

D - WRONG: Per the stem, LIC-110Y is not responding and therefore considered unavailable.

References

PLC-01-C, Rev. 3, Pg. 31 of 65

Comments and Question Modification History

NRC K/A System/E/A

System 028 Pressurizer (PZR) Level Control Malfunction

Number AA2.10 **RO** 3.3 **SRO** 3.4 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: Whether the automatic mode for PZR level control is functioning improperly, necessity of shift to manual modes

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

19 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Parent ☐ Selected for Exam ☐ Past NRC Exam? **0075164**

The plant is in normal operation at 100% power, when a second charging pump is started.||The PPO adjusts the bias thumbwheel on HIC-110 (letdown flow controller) after starting the second charging pump to maintain pressurizer level at setpoint.||Then, the second charging pump started earlier trips on overload (breaker fault).|||What is the expected plant response to the existing conditions?||

A The bias output on the letdown flow controller will automatically lower, causing letdown flow to lower and stabilize pressurizer level at setpoint. | ☐

B The pressurizer level controller will lower letdown flow and stabilize level at setpoint. | ☐

C Pressurizer level will lower until it empties or the plant trips on low pressure (TM/LP). | ☐

D Pressurizer level will lower and stabilize at some lower level below setpoint. | ☑

Justification

Pressurizer level will lower because the PZR level controller is a proportional controller and will not return level to setpoint. However, the controller output will lower to prevent the PZR from empting.
The bias on the letdown controller must be operated manually.

References

Comments and Question Modification History

NRC K/A System/E/A

System Number	RO	SRO	CFR Link

NRC K/A Generic

System Number	RO	SRO	CFR Link

20 | Origin: New | ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? | ☑ Selected for Exam ☐ Past NRC Exam? | Question ID: 6000020

The plant is at 100% power, steady state, all equipment functioning normally.

Then, the high voltage power supply to the Channel "A" Wide Range Nuclear Instruments fails, such that the channel is now reading eight (8) decades lower.

Which of the following describes the change in plant/component conditions due to this power supply failure?

A The channel indication on C04 will swap from "% Power" to "CPS". ☑

B The Power Trip Test Interlock (PTTI) for the channel is now armed. ☐

C The Dropped CEA annunciator will alarm on C04 and this channel. ☐

D The Level 1/2 Bistables are bypassing the applicable interlocks. ☐

Justification

A - CORRECT: The Wide Range NI's auto-swap .indication from "%-Power" to "CPS" when the output reaches ~ 1000 counts. If the detector is now 10E8 less sensitive, it should be reading in the source range, or CPS.

B - WRONG: The PTTI interlock would indeed be armed for this channel, IF this were the Linear Channel detector power supply that failed.

C - WRONG: The "Dropped CEA" alarm would activate on this failure, IF it were the Safety NI's not the Wide Range Detectors.

D - WRONG: The Level 1/2 bistables will activate on this channel, but ALL four channels must activate for the applicable interlocks to be affected.

References

NIS-01-C, Rev. 3, Ch. 2, Pg. 14 of 57

Comments and Question Modification History

{Per Lead Examiner, redrew K/A and generated new question - RLC}

NRC K/A System/E/A

System 032 Loss of Source Range Nuclear Instrumentation

Number AK2.01 **RO** 2.7* **SRO** 3.1 **CFR Link** (CFR 41.7 / 45.7)

Knowledge of the interrelations between the Loss of Source Range Nuclear Instrumentation and the following: Power supplies, including proper switch positions

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

20 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Org ☐ Selected for Exam ☐ Past NRC Exam? **6000014**

A reactor startup is in progress with the following existing conditions:

* All administrative requirements are fully met.
* Reactor power is being raised above the Point Of Adding Heat.
* Preparations for continuing the plant startup are on-going.

Then, a Wide Range channel fails "as-is".

Which of the following describes existing Operations Department Requirements that will ensure the failure of the Wide Range channel is recognized?

A I&C technicians are presently required to monitor proper operation of the applicable plant systems. ☐

B The RO is presently required to CONTINUOUSLY monitoring all applicable nuclear indications. ☑

C Reactor Engineering is presently required to CONTINUOUSLY monitor all reactivity indications. ☐

D A Dedicated SRO is presently required to verify proper operation of all reactor control systems. ☐

Justification

A CORRECT: OP-2202, Reactor Startup, requires the RO continuously monitor all applicable nuclear indications whenever changing reactor power.

B WRONG: I&C monitors the CEDS for proper operation during a reactor startup, NOT the Nuclear Instrumentation system. They would only be monitoring these indications during surveillance testing or calibration of these instruments, which is NOT done at this time.

C WRONG: RE is only required to monitor the NIs until the 1/M plots are < 1, or the reactor is called critical. After that time they are not required to even be in the control room.

D WRONG: The dedicated SRO is used to monitor/supervise any changes made in reactivity during the reactor startup. However, once the reactor is critical, this SRO is NOT required to CONTINUOUSLY monitor the NIs for proper response and may be monitoring other plant systems for indication of reaching the POAH.

References

Comments and Question Modification History

NRC K/A System/E/A

System 032 Loss of Source Range Nuclear Instrumentation

Number AK2.01 **RO** 2.7* **SRO** 3.1 **CFR Link** (CFR 41.7 / 45.7)

Knowledge of the interrelations between the Loss of Source Range Nuclear Instrumentation and the following: Power supplies, including proper switch positions

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

21 | Origin: ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? | Question ID: 6100044
Mod ☑ Selected for Exam ☐ Past NRC Exam?

A reactor startup is being performed with all systems and components operating as designed.

Then, with Group 2 Regulating CEAs at 70 steps withdrawn, 'A' wide range NI channel fails HIGH.

Which one of the following describes the required actions and why?

A The startup must be suspended because any further CEA withdrawal would require bypassing and making inoperable the CMI interlock. ☑

B The Startup must be suspended because continued CEA withdrawal is not allowed by Tech. Specs. with only three (3) wide range detectors. ☐

C The startup may continue once the remaining wide range detectors are verified operable by performing the applicable Tech. Spec. surveillance. ☐

D The startup may continue without pause if the remaining three (3) wide range detectors have been monitored as trending normally. ☐

Justification

A - CORRECT: The maximum CEA insertion limit is automatically bypassed when all four wide range NI channels sense reactor power to be below 1E-4%. This allows CEAs to be withdrawn for a reactor startup without the need to bypass the CMI circuit, which is required by Tech. Specs. To be operable before entering Mode 2. However, when the NI channel failed high, it caused the PDIL interlock associated with CEAPDS to become "live". Because a startup is in progress, the CEAs are not yet above the minimum PDIL (Group 4 above 72 steps) and would, therefore, immediately trigger a PDIL - CEA Motion Inhibit (CMI). The only way to continue moving CEAs in any direction at that point would be to bypass the CMI.

B - WRONG: Tech. Specs allows the reactor to be started up with only two wide range NI channels operable, but the Reactor Startup procedure requires all four NI channels be operable.

C - WRONG: Performing a Tech. Spec. surveillance on the alternate facilities to prove their operability is a requirement of some Tech. Specs. To allow continued operation, but not the one for the Wide Range Channels.

D - WRONG: This is allowed by Tech. Specs., but not by the Reactor Startup procedure, OP-2202.

References

CLP CED-01-C [Control Element Drive System]; page 32 of 67.

Comments and Question Modification History

**** Stem refers to "Group 2 Shutdown CEAs", should state "Group 2 Regulating CEAs". ****
{Fixed 'typo' in stem, "Shutdown" changed to "Regulating" - RLC}

NRC K/A System/E/A

System 033 Loss of Intermediate Range Nuclear Instrumentation

Number AK3.01 **RO** 3.2 **SRO** 3.6 **CFR Link** (CFR 41.5,41.10 / 45.6 / 45.13)

Knowledge of the reasons for the following responses as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Termination of startup following loss of intermediate- range instrumentation

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

21 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 1000044**

Parent ☐ Selected for Exam ☐ Past NRC Exam?

Which of the following indicates a loss of high voltage to the detectors for the 'A' wide range NI channel?

A Channel 'A' wide range NI meter on panel C04 drops to the bottom of the scale. ☑

B Channel 'A' wide range NI meter on panel C04 shifts to % power as indicated by "% Power" lit on the indicator above the meter. ☐

C Annunciator "CH 'A' Wide Range Extended Range CPS" on panel C04 clears. ☐

D "Non-OPR" light flashing on the Channel 'A' RPS wide range NI drawer. ☐

Justification

A: correct, although the meter would still have power the detector would not generate any pulses; B: this occurs when the extended range bistable de-energizes, examinees may chose if they believe that the channel would default to the linear detector; C: this alarm clears at 1000 cps increasing, examinees may chose if they believe that the alarm power is derived from the detector; D: all of the 'alarm' lights on the NI drawers flash when they clear, solid when they are activated, examinees may chose if they don't recall this

References

Comments and Question Modification History

NRC K/A System/E/A

System 033 Loss of Intermediate Range Nuclear Instrumentation

Number AA1.01 **RO** 2.9 **SRO** 3.1 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Power-available indicators in cabinets or equipment drawers

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

22 | Origin: New | ☑ R | ☐ SRO | ☐ Student Handout? | ☐ Lower Order? | Question ID: 6000015
☑ Selected for Exam | ☐ Past NRC Exam?

A PEO has just reported the start of a Clean Liquid Radwaste (CLR) discharge to Long Island Sound.

Five (5) minutes later, the SPO notices the Unit Two stack radiation monitor is going up.

Which one of the following system or component conditions would cause this rise in the stack radiation detector?

A The CLR Monitor Tank Pump recirculation Valve has been left open while discharging the tank. ☐

B The CLR Receiver Tank, NOT in service, is left aligned to the Monitor Tank that is discharging. ☐

C The CLR discharge filter is plugged causing the Monitor Tank Pump discharge relief valve to lift. ☐

D The CLR Monitor Tank Pump seal is leaking several gallons per minute onto the tank room floor. ☑

Justification

D - CORRECT: As the CLR system is "sealed" from the atmosphere due to entrained RCS gas, Pump seals leaking on the CLR Monitor Tank pump would "vent" contaminated gasses to the Aux. Bldg.

A - WRONG: The CLR Monitor Tank CANNOT be aligned for discharge, due to system interlocks, if the "recirc." valve is opened.

B - WRONG: The CLR Monitor Tank CANNOT be aligned for discharge, due to system interlocks, if a CLR Receiver Tank is still aligned to the Monitor Tank.

C - WRONG: The CLR discharge relief just recircs. back to the Monitor Tank, therefore should not be a source of gas release.

References

ALR-04-C, Rev. 3, Figure 1 (Rev 4)

Comments and Question Modification History

NRC K/A System/E/A

System 059 Accidental Liquid Radwaste Release

Number AK2.02 **RO** 2.7 **SRO** 2.7 **CFR Link** (CFR 41.7 / 45.7)

Knowledge of the interrelations between the Accidental Liquid Radwaste Release and the following: Radioactive-gas monitors

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

23 **Origin:** New ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? **Question ID: 6000007**
☑ Selected for Exam ☐ Past NRC Exam?

The unit is at 100% power, when an Engineering Analysis determines both Containment High Range Radiation Monitors are inoperable. The applicable Technical Specification LCO has been entered.

Due to the loss of the CTMT radiation monitors, the Unit Supervisor has directed the PPO to monitor the Main Steam Line Radiation Monitor as an alternate indication, in the event a major accident occurs causing high radiation in containment.

Which one of the following describes a limitation of using this alternate means of monitoring CTMT radiation level?

A Alternate indication will only respond if Steam Generator Tube Leakage/Rupture is present. ☐

B Alternate indication will be several decades below the real containment radiation level. ☑

C Alternate indication will only respond if the unit is at power and N-16 is present in the RCS. ☐

D Alternate indication will be unusable in the event of a Main Steam Isolation (MSI) actuation. ☐

Justification

B - CORRECT: Although the MS line Rad. Monitors are designed to monitor N-16, they would be sensitive to any form of radiation because they are simply a very sensitive radiation detector positioned in direct contact with the main steam line. However, although they are more sensitive, due to their position OUTSIDE CTMT the detectors would "see" the equivalent radiation seen by the CTMT High Range detectors as a much lower than actual radiation field. The CTMT wall would attenuate much of the radiation preventing the MS line detectors from getting a true reading.

A - WRONG: This is what the steam line detectors were DESIGNED to detect, but, based on their design, it is not the only thing they CAN detect.

C - WRONG: The MS line detectors were designed to detect N-16, but due to their design would pick up radiation from any source that strikes them..

D - WRONG: This implies the detector sensitivity is the only factor to be considered in using these detectors for the purpose stated in the stem.

References

OP-2383B, Rev. 008-04, Note on Page 6
MP2 Radiation Monitor Manual (Monman2.doc), Page 34 of 98, "Purpose of Main Steam Line radiation monitors (last sentence).

Comments and Question Modification History

D -- not plausible "Alternate indication will be unusable in the event of main steam line isolation"
[added suggestion to "D" with slight word modification - RLC]

NRC K/A System/E/A

System 061 Area Radiation Monitoring (ARM) System Alarms

Number AK1.01 **RO** 2.5* **SRO** 2.9 **CFR Link** CFR 41.8 / 41.10 / 45.3)

Knowledge of the operational implications of the following concepts as they apply to Area Radiation Monitoring (ARM) System Alarms: Detector limitations

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

24 Origin: ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? **Question ID:**
New ☑ Selected for Exam ☐ Past NRC Exam? **6000010**

The plant is at 100% power when a fire breaks out in the 25' Cable Vault.

Smoke is beginning to fill the Control Room and the crew has just noticed that the Plant Process Computer has STOPPED updating any data.
The SM then immediately directs the following crew actions per AOP-2579A, Appendix "R" Fire:

1) Trip the reactor and verify it is fully tripped.

2) Then, all personnel immediately evacuate the Control Room.

Why are these the ONLY crew actions directed to be taken before evacuating the Control Room?

A If two (2) or more CEAs do not fully insert, Emergency Boration MUST first be initiated from the Control Room. ☐

B There is NO indication that all the CEAs fully inserted outside of the Control Room, when the reactor was tripped. ☑

C If the reactor is tripped from outside the Control Room, the Main Turbine would then trip BEFORE the reactor. ☐

D There is NO way to start a backup charging pump for Emergency Boration if the running pump trips on a hot short. ☐

Justification

B - CORRECT: The ONLY way to verify the CEAs are fully inserted, without the PPC, is by the Core Mimic or CEAPDS. Both can be monitored only in the Control Room. Although it may be possible to "see" the indication through the rear door window of the Control Room, if the room were filled with smoke this would not be possible.

A - WRONG: App. "R" procedures have directions for establishing Emergency Boration from outside the Control Room.

C - WRONG: When the reactor is tripped manually, from ANY location, the Turbine is tripped automatically by the loss of power to the CEA buss undervoltage relays. App. "R" does have steps to locally trip the Main Turbine, and this could be directed before the reactor is tripped, but it would be a violation of the procedure sequence and Operational Guidelines.

D - WRONG: It is still possible to take "local" control of the Fac. 2 charging pump(s) at C-10 (Fire S/D Panel) and run it from there even if it tripped on a "hot short". There are, however, NO controls available outside the control room to start any "backup" charging pumps.

References

AOP-2579A, Rev. 9-04, Pg 6 of 64, Step 5a

Comments and Question Modification History

"A" & "D" not feasible given the stem conditions.
[Please reevaluate; I reworded "D" to say "hot short" instead of "trips on high PZR level", which is more plausible given a fire in the 25' cable vault (all the control board wiring goes through there). Also, please note that action "1)" in the stem to verify the reactor is tripped means to ensure no more than one CEA is not fully inserted. If that is not met, then Emergency Boration is required to be immediately started due to a lack of Shutdown Margin (only analyzed for ONE stuck rod). If less than two CEAs are stuck out then Emergency Boration does not have to be started for 24 hours (Xenon) or until a cooldown is commenced. It can be done from outside control, but it is a HUGE hassle, so I thought this sounded cool. - RLC]

NRC K/A System/E/A

System 067 Plant fire on site

Number AK3.02 RO 2.5 **SRO** 3.3 **CFR Link** (CFR 41.5,41.10 / 45.6 / 45.13)

Knowledge of the reasons for the following responses as they apply to the Plant Fire on Site: Steps called out in the site fire protection plan, FPS manual, and fire zone manual

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

25 Origin: ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? **Question ID:**
Mod ☑ Selected for Exam ☐ Past NRC Exam? **6100069**

The plant has tripped and EOP 2525 is being performed due to a large SGTR on #1 SG. On the trip, the RSST was lost due to grid instabilities.

All other systems responded normally with the following actions having been taken thus far:

* All initial and subsequent actions of EOP-2525 have been completed.
* The PPO has verified Natural Circulation establishing and SIAS/CIAS/EBFAS manually actuated on low RCS pressure.
* The SPO has closed both MSIVs, broke condenser vacuum and stabilized RCS temperature using both Atmospheric Dump Valves (ADVs).
* The PPO and SPO are now awaiting diagnostic queries from the US.

Then, a fuel pin suddenly ruptures due to a manufacturing defect.

Which one of the following radiation monitors would indicate the fuel failure under present plant conditions?

A Containment Refueling Bridge Area and the Containment High Range radiation monitors. ☐

B Steam Jet Air Ejector and the Steam Generator Blowdown radiation monitors. ☐

C The Main Steam Line radiation monitors for the "A" main steam header and "A" ADV. ☑

D The Main Steam Line radiation monitors for both main steam headers and both ADVs. ☐

Justification

C - CORRECT: Because the "A" ADV taps off the steam header very close to CTMT, it has it's own Rad. Monitor (unlike the "B" ADV). Ordinarily, a fuel failure would show up on several Rad. Monitors, but not with a SGTR event affecting it and a SIAS/CIAS actuation.

A - WRONG: Location of RMs inside CTMT would require significant clad failure for alarm to come in.

B - WRONG: SGBD sampling flow path was isolated on the CIAS and the SJAE has no steam supply.

D - WRONG: The "B" MSL Rad. Monitor only sees the "B" main steam header, there is no "shine" from CTMT with a SGTR and the rad monitors are UPSTREAM of where the two headers mix. Also, there is no rad. Monitor for the "B" ADV, only the "A" ADV, due to the "A"s proximity to the CTMT wall.

References

ARP-2590A-117, Rev. 000, Pg. 1 of 1

Comments and Question Modification History

NRC K/A System/E/A

System 076 High Reactor Coolant Activity

Number AK2.01 **RO** 2.6 **SRO** 3.0 **CFR Link** (CFR 41.7 / 45.7)

Knowledge of the interrelations between the High Reactor Coolant Activity and the following: Process radiation monitors

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

25 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 1000069**
Parent ☐ Selected for Exam ☐ Past NRC Exam?

The plant was tripped and EOP 2536 ESDE entered after EOP 2525 due to a large MSLB on #2 SG inside containment.
#2 SG blew down completely and was isolated as directed in the EOP.
RCS temperature and pressure are stabilized with Th subcooled margin at 94° F.
There are no indications of any fuel clad failures.

Suddenly pressurizer level and sub-cooled margin start lowering.
RCS temperatures are stable.
The STA reports that he suspects an SGTR has occurred in #2 SG.

An alarm on which of the following radmonitors would be used to confirm this diagnosis and what procedural guidance would be directed?

A Containment refueling bridge area radmonitor; transition to EOP 2540, FRP. ☑

B Steam Jet Air Ejector radmonitor; transition to EOP 2534, SGTR. ☐

C Main Steam Line radmonitor RM 4299C; remain in EOP 2536 and refer to AOP 2569, SGTL. ☐

D Steam Generator Blowdown radmonitor; repeat EOP 2525 and re-diagnose the event. ☐

Justification

A - CORRECT: with low RCS activity and the ruptured SG already faulted this RM and the personnel access hatch area RM are the only RMs capable of alarming (SJAE & SGBD RMs isolated by MSIS/CIAS and no steam flow for RM4299C), 2 simultaneous events requires FRP IAW 2260;

B - WRONG: MSIVs are closed, no pathway exists;

C - WRONG: location of RM and 30 mr/hr alarm setpoint would require significant clad failure for alarm to come in;

D - WRONG: SGBD sampling was isolated during SG isolation and/or CIAS, no pathway

References

Comments and Question Modification History

NRC K/A System/E/A

System 2.4 Emergency Procedure /Plan

Number G **RO** **SRO** **CFR Link**

SEE GENERIC K/A

NRC K/A Generic

System 2.4 Emergency Procedures /Plan

Number 2.4.46 **RO** 3.5 **SRO** 3.6 **CFR Link** (CFR: 43.5 / 45.3 / 45.12)

Ability to verify that the alarms are consistent with the plant conditions.

26 **Origin:** New ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? **Question ID: 6000006**
☑ Selected for Exam ☐ Past NRC Exam?

The plant tripped from 100%. On the trip both Bus 25A and Bus 25B failed to transfer to the RSST.

The difference between Thot and CET temperatures is found to be 15°F.

Which one of the following describes the operator actions that must be taken to mitigate this condition in accordance with EOP 2528, Loss of Offsite Power/Loss of Forced Circulation?

A Stop any cooldown and ensure both steam generator saturation temperatures are within 10°F. ☐

B Energize all available Pressurizer heaters and ensure RCS subcooled per highest Thot. ☐

C Stop any RCS depressurization and begin venting the reactor head using the installed head vents. ☐

D Ensure steam generator (SG) pressure is less than saturation pressure for existing Tcold by steaming the SGs. ☑

Justification

D - CORRECT: This is the guidance given in the procedure, as it indicates not enough heat is being withdrawn from the RCS by the SGs. This will soon result in stalled NC flow, indicative of the larger than expected delta-T between CETs and Thot.

A - WRONG: This is the course of action when the two RCS loops become "uncoupled" due to uneven steam generator heat removal.

B - WRONG: This implies the abnormal delta-T is due to void formation, but the void would have to reach the top of the core and effect both detectors.

C - WRONG: This implies noncondensible gasses have come out of solution and stagnated RCS flow, resulting in the abnormal delta-T.

References

EOP-2528, Rev. 15,Pg. 8 of 36, Step 11, Contingency

Comments and Question Modification History

D add "by steaming steam generators".
[done - RLC]
{Reworded "C" - RLC}

NRC K/A System/E/A

System A13 Natural Circulation Operations

Number AK1.3 **RO** 3.1 **SRO** 3.4 **CFR Link** (CFR: 41.8 / 41.10 / 45.3)

Knowledge of the operational implications of annunciators and conditions indicating signals and remedial actions associated with the Natural Circulation Operations as they apply to the Natural Circulation Operations.

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

27 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? **Question ID:**
Bank ☑ Selected for Exam ☐ Past NRC Exam? **6073999**

A plant startup is in progress at End-Of-Life following an uncomplicated plant trip. Power is being held at 1% while operators prepare to start the first Main Feedwater Pump.

Then, a major plant accident occurs with the following conditions being noted

- "A" Steam Generator level is 200" and dropping rapidly.
- "B" Steam Generator level is 15% and dropping slowly.
- "A" Steam Generator pressure is 350 psia and dropping rapidly.
- "B" Steam Generator pressure is 740 psia and dropping slowly.
- Pressurizer pressure is 1380 psia and dropping.
- Pressurizer level is off scale low.
- RCS temperature is 460°F and dropping.
- CTMT pressure is 28 psig and rising.
- No rad monitors are rising or in alarm.

Which one of the following describe administrative limits that are in place to prevent this accident from exceeding design bases?

A The Moderator Temperature Coefficient has a maximum, allowable negative limit when critical. ☑

B The steam generators have a maximum allowable water level when operating in Mode 2. ☐

C The combination of ASI, Tcold, Q-power and RCS pressure are within limits above the POAH. ☐

D The maximum amount of CEA insertion allowable for the given power level of the reactor. ☐

Justification

A - CORRECT: Limiting the maximum negative value of MTC ensures the safety analysis remains valid for limiting the return to power following an ESD event and the subsequent cooldown.

B - WRONG: Although the amount of mass in the SG will certainly have an impact on the magnitude of the ESD event, and the maximum allowable SG water level is administratively limited by plant procedures, this limit is based on protecting plant systems and equipment.

C - WRONG: This basically describes the TM/LP limit function, which, although certainly affected by an ESD, is not designed to protect the plant/core in such an event.

D - WRONG: This defines the PDIL, which will be affected as plant power changes with the ESD. Also, the PDIL is one of the foundation assumptions used in the design of various LSSS. However, the bases for the PDIL does not encompass any plant transients bounded by an ESD event.

References

Tech. Spec. Bases 3/4, Page 1-1; Shutdown Margin (ITC)
Tech. Spec. Bases 3/4, Page 1-1a; MTC
CLP E36-01-C; Page 9 of 33; Subsection 2.j
CLP E36-01-C; Figure 4 - Safety Analysis Assumptions
CLP E36-01-C; Figure 7 - FSAR Worse Case Analysis

Comments and Question Modification History

NRC K/A System/E/A

System A11 RCS Overcooling

Number AA2.2 **RO** 3.0 **SRO** 3.4 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret adherence to appropriate procedures and operation within the limitations in the facility's license and amendments as they apply to the RCS Overcooling.

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

28 | Origin: New | ☑ R | ☐ SRO | ☐ Student Handout? | ☑ Lower Order? | Question ID: 6000110
☑ Selected for Exam | ☐ Past NRC Exam?

A plant heatup is in progress in accordance OP 2201, "Plant Heatup". SDC is in operation and the crew is ready to rack up the breakers for "A" and "B" RCPs to commence concurrent RCP/SDC operation.

The current RBCCW valve alignment is:

2-RB-30.1A, RBCCW CTMT ISOL HDR A SPLY - CLOSED
2-RB-37.2A, RBCCW CTMT ISOL HDR A RTN - OPEN
2-RB-30.1B, RBCCW CTMT ISOL HDR B SPLY - CLOSED
2-RB-37.2B, RBCCW CTMT ISOL HDR B RTN - OPEN

Which of following describes the system response with this valve alignment?

A Annunciator "RCP B CLG WTR FLOW LO, would clear when the "A" RCP breaker is racked-up and the "B" RCP can be started. ☐

B Annunciator "RCP A CLG WTR FLOW LO, would NOT clear when the "A" RCP breaker is racked-up and the "A" RCP can NOT be started. ☑

C Annunciator "RCP A CLG WTR FLOW LO, would clear when the "A" RCP breaker is racked-up and the "A" RCP can NOT be started. ☐

D Annunciator "RCP B CLG WTR FLOW LO, would NOT clear when the "A" RCP breaker is racked-up, but the "B" RCP can be started. ☐

Justification

B - CORRECT: Since the RBCCW supply to "A" is isolated because 2-RB-30.1A is closed, the low flow annunciator will not clear when the breaker is racked-up and the RCP will not start because one of the start permissive for the RCP is adequate RBCCW flow

A - WRONG: The low flow annunciator will not clear because there is no RBCCW flow to "A" RCP and the RCP cannot be started because the start permissive is not met.

C - WRONG: The low flow annunciator will not clear because there is no flow to "A" RCP.

D - WRONG: The "A" RCP can not be started because the RBCCW flow permissive is not meet.

References

ARP 2590B-071, Rev. 000 for "RCP A CLG WTR FLOW LO" (C-02/3 Window DA-17) Corrective Action Step 4.

Comments and Question Modification History

TOO SIMPLE
[Modified - SMD] Approved - Gil Johnson
{Modified stem per suggestion - RLC}

NRC K/A System/E/A

System 003 Reactor Coolant Pump System (RCPS)

Number K6.04 **RO** 2.8 **SRO** 3.1 **CFR Link** (CFR: 41.7 / 45/5)

Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: Containment isolation valves affecting RCP operation

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

29 | **Origin:** New | ☑ R | ☐ SRO | ☐ Student Handout? | ☐ Lower Order? | **Question ID:** 6000013
☑ Selected for Exam | ☐ Past NRC Exam?

The plant is at 100% power, steady state. A ground exists somewhere in the Letdown Flow Control circuit causing continuous letdown flow oscillations. Letdown Flow Control Valve 2-CH-110Q is being placed in service and 2-CH-110P is being swap out in hopes of isolating the source of the ground. The control operator has NOT had a chance to fully restore CVCS controls to their normal, at power, alignment.

Then, immediately after operators complete unisolating 2-CH-110Q, the now in-service CH-110Q fails OPEN, resulting in the following conditions:

- Letdown flow is 128 gpm.
- Letdown Temperature through the Ion Exchangers (I.X.'s) is 150 °F.
- Letdown Pressure upstream of the Pressure Control Valves is oscillating between 300 - 350 psi.

Which of the following describes the plant or system response, and the required actions based on that response, to CH-110Q failing open at this time.

A Letdown Flow is exceeding the design limit of the Ion Exchangers; they must be verified bypassed before the resin is channeled. ☐

B Letdown Pressure is oscillating above the letdown design limit; the backpressure controller must be stabilized in manual to close the relief valve. ☐

C Letdown flow is exceeding the capacity of the Letdown Heat Exchanger; ensure the I.X.'s are bypassed to prevent I.X. damage on temperature. ☑

D Letdown flow is exceeding the capacity of the running charging pumps; the plant must be tripped due to inability to maintain PZR level. ☐

Justification

C - CORRECT: L/D flow may be exceeding the capacity of the HX. The auto system function should bypass the IX on high L/D temp., preventing IX damage on temperature, IF the "bypass" switch has been returned to "auto". Based on the conditions given in the stem, the control operator must not have yet returned the switch to auto and the I.X.'s are NOT being bypassed. Therefore, they are in immediate danger of damage on high temperature.

A - WRONG: The Letdown Limiter circuit is designed to ensure letdown flow does not exceed the IX capacity by limiting the amount of "open" signal the normal control system can send to the Letdown Flow Control valves. However, even if the flow control valves FAIL open, the letdown valves are "mechanically blocked" to prevent letdown flow from exceeding 128 gpm.

B - WRONG: Pressure may INITIALLY lift the relief, but a backpressure of ~350psi is within the design limit of the letdown piping because the relief does not lift to over 500 psi. The back-pressure control valves are operating to limit the magnitude of the pressure rise due to the high letdown flow. No operator actions are required for this condition.

D - WRONG: Even with the L/D control valve failed full open, L/D flow is not designed to exceed the capacity of all three charging pumps with normal RCS pressure. The backup charging pumps will automatically start on PZR level and maintain it within limits without the need to trip the plant.

References

CVS-00-C, (CVCS Training Material) Rev. 8, Ch. 3, Pages 2 & 29

Comments and Question Modification History

{Chaged 'C' to say "ensure I.X.'s are bypassed" vs. ""the I.X.'s must be bypassed" (which incorrectly implies the 'auto bypass' has failed - RLC}

NRC K/A System/E/A

System 004 Chemical and Volume Control System

Number A2.21 **RO** 2.7 **SRO** 2.7 **CFR Link** (CFR: 41.5/ 43/5 / 45/3 / 45/5)

Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Excessive letdown flow, pressure, and temperatures on ion exchange resins (also causes)

NRC K/A Generic

29 Origin: New ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 6000013**

☑ Selected for Exam ☐ Past NRC Exam?

System Number RO SRO CFR Link

29 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Org ☐ Selected for Exam ☐ Past NRC Exam? **6000013**

The plant is at 100% power, steady state, Letdown Flow Control Valve 2-CH-110Q has just been placed in service due to a packing leak on 2-CH-110P. All CVCS controls are in normal, at power, alignment.

Then, five minutes after operators complete isolating 2-CH-110P, the now in-service CH-110Q fails full OPEN. Letdown flow is 130 gpm.

Which of the following describes the plant or system response, and the required actions based on that response, to CH-110Q failing opened at this time.

A Letdown Flow would exceed the design limit of the Ion Exchangers; they must be verified bypassed before the resin is channeled. ☑

B Letdown Pressure would exceed the letdown relief valve setpoint; the backpressure controller must be stabilized in manual to close the relief valve. ☐

C Letdown flow would exceed the capacity of the Letdown Heat Exchanger; letdown must be secured to prevent I.X. damage on temperature. ☐

D Letdown flow would exceed the capacity of the running charging pump; ensure both backup charging pumps are running in manual. ☐

Justification

A - CORRECT: The Letdown Limiter circuit is designed to ensure letdown flow does not exceed the IX capacity by limiting the amount of "open" signal the normal control system can send to the Letdown Flow Control valves. However, if the flow control valves FAIL open, the circuit is of no consequence.

B - WRONG: Pressure would INITIALLY lift the relief, but the Backpressure controller is designed to respond in auto to open the back-pressure control valves and limit the magnitude and duration of the pressure spike. No operator actions are required for this function.

C - WRONG: L/D flow may exceed the capacity of the HX. However, auto system functions will bypass the IX on high L/D temp., preventing IX damage on temperature. No operator action required.

D - WRONG: Even with the L/D control valve failed full open, L/D flow is not designed to exceed the capacity of all three charging pumps with normal RCS pressure. The backup charging pumps will automatically start on PZR level without placing the system in manual control.

References

CVC-00-C, Rev. 8, Ch. 3, Pg. 10 of 165

Comments and Question Modification History

Answer "A" check channeling as proper for high flow condition.
[will check with SME this week, however, this is what I have heard at MP2 since I got my first License back in '84 - RLC]
{Rewrote question based on validation input - RLC}

NRC K/A System/E/A

System 004 Chemical and Volume Control System

Number A2.21 **RO** 2.7 **SRO** 2.7 **CFR Link** (CFR: 41.5/ 43/5 / 45/3 / 45/5)

Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Excessive letdown flow, pressure, and temperatures on ion exchange resins (also causes)

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

30 Origin: New ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 6000016**
☑ Selected for Exam ☐ Past NRC Exam?

The following plant conditions exist:

* 20 gpm RCS leak into CTMT has recently appeared.
* Control room operators are performing a Rapid Downpower at 30%/hr.
* Plant power is presently 93% and dropping at the intended rate.
* 2 charging pumps are running, Letdown is at approximately 60 gpm.
* Pressurizer level is 65% and stable.
* VCT level is 75% and stable.
* Boric Acid addition to the Charging pump suction at the required rate.
* Additional 20 gpm blended makeup to the Charging pump suction.
* A plant Chemist is regenerating a CPF Ion Exchanger resin bed.

Then, during the load reduction, the standby PMW pump trips off line on overload, resulting in a PMW Low Pressure alarm.

Based on the above plant conditions and the design limits of the CVCS and it's support systems, how will the plant respond to the PMW pump trip?

A The PMW and Boric Acid Flow Controllers, both operating in automatic, will maintain BOTH flow rates at setpoint. ☐

B Pressurizer level will begin to drop BELOW Level Setpoint, with all RCS temperatures being maintained on program. ☐

C The RATE of the plant downpower will accelerate without any operator action on ANY reactivity control systems. ☑

D Boric Acid and PMW flow rates will BOTH drop, maintaining the preset blend concentration to the charging pumps. ☐

Justification

C - CORRECT: With PMW pump tripped and chemistry performing a regen., PMW pressure will drop enough to prevent PMW flow to the CVCS

A - WRONG: The flow controller cannot make up for the loss of PMW pressure.

B - WRONG: RCS temperatures will not be maintained.

D - WRONG: ONLY the PMW flow rate will drop, not BOTH.

References

ARP-2590B-020 Alarm D-5, NOTE 2

Comments and Question Modification History

NRC K/A System/E/A

System 004 Chemical and Volume Control System

Number A1.10 **RO** 3.7 **SRO** 3.9 **CFR Link** (CFR: 41.5/45.5)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CVCS controls including: Reactor power

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

31 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? **Question ID:**
Mod ☑ Selected for Exam ☐ Past NRC Exam? **6075321**

When entering 'Reduced Inventory Region' (RIR), SDC flow is reduced to between 1400 and 1600 gpm.

What is the reason for the MINIMUM, REQUIRED flow of 1400 gpm?

A This is the minimum flow required to prevent thermal stratification in the RCS hot and cold legs. ☐

B Lower flow rates have the potential to cause RCS heatup due to inadequate decay heat removal. ☐

C This reduced flow rate is consistent with the minimum required for adequate boron mixing in the smaller RCS volume. ☑

D This is the minimum required flow for the LPSI (SDC) Pump, lower flow rates would potentially overheat the pump. ☐

Justification

C - CORRECT: Due to the reduced volume in the reactor, a lower flow rate is adequate.

A - WRONG: Thermal stratification will not occur with the RCS in this configuration.

B - WRONG: Flow rates below 1000 gallons/minute would be required before inadequate decay heat removal became an issue.

D - WRONG: This is not the minimum flow for LPSI pump.

References

OP-2310, Rev. 23-00, page 8 of 107, step 3.9

Comments and Question Modification History

{Replaced "A" with suggestion from Joe D., however, comment on distractor "C" not being a plausible distractor is puzzling as this is the CORRECT answer to the question. - RLC}

NRC K/A System/E/A

System 005 Residual Heat Removal System (RHRS)

Number K5.09 **RO** 3.2 **SRO** 3.4 **CFR Link** (CFR: 41.5 / 45.7)

Knowledge of the operational implications of the following concepts as they apply the RHRS: Dilution and boration considerations

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

31 | Origin: ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? | Question ID: 0075321
Parent ☐ Selected for Exam ☐ Past NRC Exam?

When entering 'Reduced Inventory Region' (RIR), SDC flow is reduced to <1500 gpm.

What is the reason for reducing flow?

A This flow rate ensures that adequate NPSH for a LPSI pump will be maintained even if level lowers to the bottom of the hot leg. ☐

B Higher flow rates have the potential to cause crud bursts and airborne radioactivity problems. ☐

C This reduced flow rate is consistent with the reduced times for adequate boron mixing in the smaller RCS volume. ☐

D The reduced flow rate reduces the potential for air entrainment (vortexing) in the SDC suction. ☑

Justification

Vortexing has occurred at RIR levels with high SDC flows causing loss of SDC due to the suction loop high point.

References

Comments and Question Modification History

NRC K/A System/E/A

System Number	RO	SRO	CFR Link

NRC K/A Generic

System Number	RO	SRO	CFR Link

32 **Origin:** Mod ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? **Question ID:** 6071905
☑ Selected for Exam ☐ Past NRC Exam?

The plant has experienced a Loss of Coolant Accident and the RWST level is approaching the SRAS setpoint.

Which of the following valves REQUIRE operator action to ensure they go closed when SRAS fully actuates?

A RWST header isolation valves (2-CS-13.1A & 2-CS-13.1B). ☐

B Both CTMT Sump Outlet isolation valves (2-CS-16.1A & CS-16.1B). ☐

C 2-SI-659 & 2-SI-660, Safety Injection Pump Recirc. Valves. ☑

D RBCCW outlet isolation valves from the SDC heat exchangers. ☐

Justification

C - CORRECT: Components 2-SI-659 & 2-SI-660 require the switches on C-01 be placed in the

A - WRONG: The header isolations due not get a closed signal as any remaining pressure in CTMT, and head of water in the CTMT suction line, will seat the check valves in this line and make it unnecessary.

B - WRONG: CTMT Sump outlet isolations get an OPEN signal, and they get it without operator action.

D - WRONG: These valves get an OPEN signal on a SRAS and should not be closed. However, in all other ESAS situations these valves would be closed (a common misconception).

References

EOP-2532, Rev. 24, Step 21 & 48.

Comments and Question Modification History

Repair "A" to be correct
[actually, "C" is the correct answer, but, I did reword stem to ask "which valves need help to go closed", eliminating the need for it in each answer. Also I corrected the "Justification" and edited "A" & "C" as I initially intended them to be (sorry, my initial editing some how did not "take") - RLC]

NRC K/A System/E/A

System 006 Emergency Core Cooling System (ECCS)

Number A4.02 **RO** 4.0* **SRO** 3.8 **CFR Link** (CFR: 41.7 / 45.5 to 45.8)

Ability to manually operate and/or monitor in the control room: Valves

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

32 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 0071905**
Parent ☐ Selected for Exam ☐ Past NRC Exam?

The plant has experienced a Loss of Coolant Accident and the following conditions exist:||- Sump Recirculation has occurred.|- The Safety Injection Recirculation Header Isolation valves, 2-SI-659 and 660, are the ONLY SRAS actuated components that have NOT automatically positioned.||Which one of the following statements describes WHEN these valves should be closed?||

A Immediately after other SRAS actuations have been verified. | ☑

B Only after verifying 50 gpm flow from each High Pressure Safety Injection (HPSI) pump. | ☐

C Only after RWST header isolation valves (2-CS-13.1A & 2-CS-13.1B) are closed. | ☐

D Immediately after closing RBCCW outlet isolation valves from the SDC heat exchangers. | ☐

Justification

Components 2-SI-659 & 2-SI-660 are verified closed before all of the actions mentioned in the distractors. These valves being open also violate CTMT integrity and offer a direct release path from CTMT to the environment. Therefore, they should be closed as soon as they are found open.

References

Comments and Question Modification History

NRC K/A System/E/A

System Number	RO	SRO	CFR Link

NRC K/A Generic

System Number	RO	SRO	CFR Link

 # 33 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? **Question ID:**
Bank ☑ Selected for Exam ☐ Past NRC Exam? **6071028**

When drawing a bubble in the Pressurizer, which of the following sets of conditions indicates that a bubble has been formed?

A Pressurizer level lowering with constant or rising pressure.
Quench tank level, temperature and pressure rising with no water being added to the RCS. ☐

B Pressurizer level lowering with constant or rising pressure.
Pressurizer temperature corresponds to saturation temperature for present RCS pressure. ☑

C Pressurizer pressure rising with rising water space temperature.
Quench tank level, temperature and pressure rising with no water being added to the RCS. ☐

D Pressurizer pressure rising while maintaining constant charging flow.
Pressurizer steam and water space temperatures rising above hot leg temperature. ☐

Justification

B - CORRECT: OP 2301D states, "Monitor for spontaneous bubble formation by the following:
- Pressurizer level lowering with constant or rising pressure.
- Pressurizer temperature corresponds to saturation temperature for present RCS pressure.

A - WRONG: PORVs are closed as soon as the PZR indicates filled and vented by way of flooding over into the Quench Tank. Gasses that come out of solution upon heating will be vented out the PZR steam space sample line, which remains open. Leaving the PORVs open would challenge the Quench Tank rupture disk.

C - WRONG: This is indicative of a solid RCS, but does occur to some extent when drawing a bubble. However, it can NOT be used as confirmation that a bubble EXISTS in the PZR. Also, the PORVs are closed as soon as the PZR indicates filled and vented, so QT parameters should NOT change.

D - WRONG: This will occur as soon as the process for drawing a bubble in the PZR begins and the PORVs are closed. Also, PZR water space temperature must be at SATURATION for the RCS pressure, NOT just above Thot.

References

OP-2301D, Rev. 27-02, Pg. 44 of 83, Step 4.4.27

Comments and Question Modification History

NRC K/A System/E/A

System 007 Pressurizer Relief Tank/Quench Tank System (PRTS)

Number K5.02 **RO** 3.1 **SRO** 3.4 **CFR Link** (CFR:41.5/45.7)

Knowledge of the operational implications of the following concepts as the apply to PRTS: Method of forming a steam bubble in the PZR

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

34 | Origin: New | ☑ R | ☐ SRO | ☐ Student Handout? | ☑ Lower Order? | Question ID: 6000018
☑ Selected for Exam | ☐ Past NRC Exam?

Which of the following will cause the "RBCCW SURGE TK AUTO MAKEUP" annunciator to alarm on C-06/7?

A Makeup valve open for 3 minutes and makeup valve in "AUTO" ☑

B As soon as the makeup valve opens, regardless of PMW header pressure. ☐

C Only when the setpoint for "RBCCW SURGE TK Level HI/LO" setpoint is reached. ☐

D When makeup valve opens, if PMW header pressure is insufficient to supply makeup. ☐

Justification

A - CORRECT: The annunciator setpoint is "valve open for 3 minutes".

B - WRONG; There is no alarm function from the valve position.

C - WRONG; The surge HI/LO alarm is not tied to the SURGE TK excessive makeup alarm.

D - WRONG; Valve position and PMW pressure do not feed into this alarm.

References

ARP-2590E, Rev. 009-01, B-8, 046

Comments and Question Modification History

NRC K/A System/E/A

System 008 Component Cooling Water System (CCWS)

Number K4.02 **RO** 2.9 **SRO** 2.7 **CFR Link** (CFR: 41.7)

Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: Operation of the surge tank, including the associated valves and controls

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

35 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Mod ☑ Selected for Exam ☐ Past NRC Exam? **6069977**

The plant is in Mode 5 on Shutdown Cooling. The crew is presently making preparations to perform PORV Block Valve stroke testing. The test involves cycling each Block Valve fully closed then fully open.

Which of the following conditions meet the system requirements that would allow stroke testing of the PORV Block Valves?

	RCS Pressure	RCS Tc	LTOP Switch Position	SDC Status
#1)	250 psia	150 °F	Low	In Service
#2)	250 psia	150 °F	High	In Service
#3)	300 psia	175 °F	High	In Service
#4)	500 psia	175 °F	Low	Secured

A #1 ☐

B #2 ☑

C #3 ☐

D #4 ☐

Justification

B - CORRECT: Conditions are within the SDC System limits, below LTOP limits (LTOP ACTUATION > 410 PSIA & < 275 DEG F) and allow the PORV Block Valve to be closed (LTOP Switch MUST be in "Low").

A - WRONG: The LTOP switch in "Low" prevents closing of the PORV Block Valve before opening the PORV, which would blow the PRT rupture disk.

C - WRONG: 300 PSIA is above the design pressure for the SDC System.

D - WRONG: 500 PSIA and 175 °F is above the LTOP limit and would open the PORV as soon as the LTOP switch was selected to "Low".

References

RCS-00-C, Rev. 7, Ch. 1, Pg 34 of 116
RCS-00-C, Figure 14 (Rev. 1)

Comments and Question Modification History

**** Rewrite to correct plant conditions for PORV/Block Valve testing. Utilize "Block Valve" testing as PORV testing has limited criteria and is non-discriminating. (Mode 5 on SDC) ****
{Reworded stem to test more complex concepts solicited from Block Valve testing and reworded Choice "B" to be correct for conditions. - RLC}

NRC K/A System/E/A

System 010 Pressurizer Pressure Control System (PZR PCS)

Number A3.01 **RO** 3.0 **SRO** 3.2 **CFR Link** (CFR: 41.7 / 45.5)

Ability to monitor automatic operation of the PZR PCS, including: PRT temperature and pressure during PORV testing

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

35 | Origin: ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? | Question ID:
Org ☐ Selected for Exam ☐ Past NRC Exam? | 6069977

A plant heat-up is progress using the "A" and "B" RCPs. The crew is presently making preparations to perform PORV stroke testing. The test involves temporarily closing the applicable Block Valve, then manually opening and closing each PORVs.

Which of the following conditions meet the system requirements that would allow manual stroke testing of the PORVs?

	RCS Pressure	RCS Tc	LTOP Switch Position	SDC Status
#1)	250 psia	150 °F	Low	In Service
#2)	250 psia	150 °F	High	In Service
#3)	500 psia	275 °F	High	Secured
#4)	600 psia	375 °F	High	Secured

A #1 ☐

B #2 ☑

C #3 ☐

D #4 ☐

Justification

B - CORRECT: Conditions are within the SDC System limits, below LTOP limits (LTOP ACTUATION > 410 PSIA & < 275 DEG F) and allow the PORV Block Valve to be closed (LTOP Switch MUST be in "Low").

A - WRONG: The LTOP switch in "Low" prevents closing of the PORV Block Valve before opening the PORV, which would blow the PRT rupture disk.

C - WRONG: 500 PSIA is above the LTOP setpoint for this temperature.

D - WRONG: 600 PSIA and 375 °F would over-pressurizes the Quench Tank (PRT) and blows the Quench Tank [PRT] rupture disk.

References

RCS-00-C, Rev. 7, Ch. 1, Pg 34 of 116
RCS-00-C, Figure 14 (Rev. 1)

Comments and Question Modification History

**** Rewrite to correct plant conditions for PORV/Block Valve testing (Mode 5 on SDC) ****

NRC K/A System/E/A

System 010 Pressurizer Pressure Control System (PZR PCS)

Number A3.01 **RO** 3.0 **SRO** 3.2 **CFR Link** (CFR: 41.7 / 45.5)

Ability to monitor automatic operation of the PZR PCS, including: PRT temperature and pressure during PORV testing

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

35 | Origin: ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? | Question ID: 0069977
Parent ☐ Selected for Exam ☐ Past NRC Exam?

Operations is performing a plant heat-up.||"A & B" RCPs are running.

Which of the following conditions comply with procedures?

	RCS Pressure	RCS Tc	LTOP Switch Position	SDC Status	
A	250 psia	150 °F	Low	In Service	☑
B	300 psia	200 °F	Low	In Service	☐
C	500 psia	250 °F	Low	Secured	☐
D	1600 psia	375 °F	High	Secured	☐

Justification

The correct answer is 'A' because it is above the NPSH for "A&B" RCPs, within the SDC curve and below LTOP limits.

References

Comments and Question Modification History

NRC K/A System/E/A

System Number	RO	SRO	CFR Link

NRC K/A Generic

System Number	RO	SRO	CFR Link

36 | Origin: Mod | ☑ R | ☐ SRO | ☐ Student Handout? | ☐ Lower Order? | Question ID: 6054501
☑ Selected for Exam | ☐ Past NRC Exam?

The Main Turbine has just tripped from 100% power due to a failure in the Electro-Hydraulic Control cabinet. The turbine trip results in RCS pressure spiking above 2410 psia before the reactor is shutdown/tripped automatically. RCS pressure then immediately dropped to a normal "post-trip" value.

Which of the following is an indication that RPS failed to trip the reactor, but that it tripped due to actuation of the Diverse Scram System (DSS)?

A The MG sets are still running but their output contactors are tripped open. ☑

B Both MG set 480 volt supply breakers are open with amber indicating lights. ☐

C 3 minutes and 20 seconds after the trip, both channels of AFAS will actuate. ☐

D All four (4) of the CEDS Bus Undervoltage Relays have been deenergized. ☐

Justification

A - CORRECT: With the RPS High Pressure trip set ~ 140 psi above normal operating pressure, the plant should have tripped on High Pressure. If it tripped on activation of the DSS, then the MG set output contactors would be open. This does NOT trip the MG sets, just deenergizes the CEA buses.

B - WRONG: The DSS does NOT trip MG set input power, just the output power.

C - WRONG: The DSS will actuate the AFAS if the high pressure signal stays in for the ENTIRE time (3 minutes, 20 seconds).

D - WRONG: The CEDS Undervoltage relays will be deenergized on the loss of power to the CEA buses, but this happens regardless of HOW the plant was tripped. Therefore it can NOT be used as an indication of the plant tripping on DSS actuation.

References

CED-00-C, Control Rod Drive System (Training Material), Rev. 3, Ch. 4, Page 9 of 67
AFW-00-C, Aux. Feedwater (Training Mat.), Rev. 5, Ch. 3, Pages 11 & 12 of 57

Comments and Question Modification History

{Modified original question (now "Mod" not "Bank") to improve K/A alignment. - RLC}

NRC K/A System/E/A

System 012 Reactor Protection System

Number K3.01 **RO** 3.9 **SRO** 4.0 **CFR Link** (CFR: 41.7 / 45.6)

Knowledge of the effect that a loss or malfunction of the RPS will have on the following: CRDS

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

36 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 0054501**

Parent ☐ Selected for Exam ☐ Past NRC Exam?

The Main Turbine has just tripped from 100% power due to a failure in the Electro-Hydraulic Control cabinet. The Diverse Scram System (DSS) then tripped the reactor by opening the CEDS MG set output contactors.

Other than the Turbine Trip actuation, what other RPS trip has failed?
(NOTE: Assume that the RPS is set MORE conservatively than the DSS trip setpoint and no other system failures have occurred)

A High Pressurizer Pressure. ☑

B High SG Level ☐

C TM/LP. ☐

D CEDS Bus Undervoltage. ☐

Justification

A - CORRECT; The load reject trip will cause a sudden rise in RCS temperature. For every degree rise in Tavg, PZR level will rise one percent. For every percent rise in PZR level, RCS pressure will rise 15 psi. With the RPS High Pressure trip set ~ 140 psi above normal operating pressure, the plant should have tripped on High Pressure.

B - WRONG; With a load reject trip, SG pressure will rise suddenly. This will result in a sudden "shrink" in SG level, not a "swell", and still not before RCS pressure rises to the trip setpoint on RPS.

C - WRONG; TM/LP setpoint will rise with RCS temperature, but not faster than RCS pressure. Although the trip setpoint will easily reach NOP, the RCS will no longer be at NOP due to the temperature rise.

D - WRONG; The CEDS Undervoltage relays trip the TURBINE on a REACTOR trip, not the other way around.

References

ARP-2590C-101, Rev. 0, Pg. 1 of 1

Comments and Question Modification History

NRC K/A System/E/A

System 012 Reactor Protection System

Number K3.01 **RO** 3.9 **SRO** 4.0 **CFR Link** (CFR: 41.7 / 45.6)

Knowledge of the effect that a loss or malfunction of the RPS will have on the following: CRDS

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

37 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Mod ☑ Selected for Exam ☐ Past NRC Exam? **6054051**

The following plant conditions exist:

- Plant is at 100% power, steady state.
- Vital 120 VAC electrical bus VIAC-3 (VA-30) is aligned to its alternate power supply.
- While in this alignment a complete Loss of Normal Power occurs CONCURRENT with a loss of Inverter 4 (INV-4 is off-line).

During the performance of EOP-2525, the STA reports the following:

- RPS channels "C" and "D" momentarily deenergized until the Diesel Generators loaded.
- All parameters monitored by safety channels "C" and "D", EXCEPT for RWST level indications, momentarily failed until the Diesel Generators loaded.
- Both facilities of the ESAS fully actuated, with the EXCEPTION of AEAS and SRAS.

Which of the following describes why a SRAS actuation did NOT occur with the rest of the ESAS actuations?

A The ESAS SRAS actuation only triggers if a High CTMT pressure SIAS actuation is also sensed by the ESAS Sensor Cabinets. ☐

B The RWST level channels have a backup power supply directly from non-vital 125 VDC, which auto-swaps on loss of the VIAC. ☐

C The RWST level channels are powered from the ESAS Sensor Cabinets and each Sensor Cabinet has a backup power supply. ☑

D When power is restored to ESAS, the "crow bar" circuit blows the Actuation Cabinet fuses before ESAS can actuate a SRAS. ☐

Justification

C - CORRECT: All ESAS sensor channels have backup power from the opposite facility. Channel 'A' from VA-10/ 40, Channel 'B' from VA-20/30, Channel 'C' from VA-30/20 and Channel 'D' from VA-40/10. Because the RWST level loops get their power from the ESAS sensor cabinets, which never totally lost power, SRAS does not fire.

A - WRONG:

B - WRONG: When the backup power to VA-30 & VA-40 (VR-11 & 21) momentarily deenergized on the LNP, ESAS saw two channels of safety parameters fail to their accident condition. This meets the 2/4 requirement and will result in an actuation of ALL systems except for SRAS and AEAS. Channel "Y" of PZR Pressure Control has the aforementioned backup DC power supply, not the RWST indication.

D - WRONG: The "crow bar" circuit will blow the Actuation Cabinet fuses IF the Actuation Cabinets lost power and then were reenergized. However ESAS Actuation Cabinets are powered from VIAC-10 and -20, which did NOT deenergize in the question scenario.

References

ESA-01-C, Rev. 3, Pg. 8 & 39 of 73

Comments and Question Modification History

NRC K/A System/E/A

System 013 Engineered Safety Features Actuation System (ESFAS)

Number K2.01 **RO** 3.6* **SRO** 3.8 **CFR Link** (CFR: 41.7)

Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

37 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Parent ☐ Selected for Exam ☐ Past NRC Exam? **0054051**

The following plant conditions exist:

- Plant is at 100% power, steady state.
- Vital 120 VAC electrical bus VIAC-3 (VA-30) is aligned to its alternate power supply.
- While in this alignment a complete Loss of Normal Power occurs CONCURRENT with a loss of Inverter 4 (INV-4 is off-line).

During the performance of EOP-2525, the STA reports the following:

- RPS channels "C" and "D" momentarily deenergized until the Diesel Generators loaded.
- All parameters monitored by safety channels "C" and "D" (CTMT press., PZR press., SG level & press., etc.) momentarily failed until the Diesel Generators loaded.

Which of the following describes the effects of this electrical alignment/casualty on the Engineered Safeguards Actuation System (ESAS)?

A A total actuation of all ESAS components and systems has occurred. ☐

B "C" and "D" ESAS sensor cabinets momentarily deenergized and then reenergized. ☐

C Both facilities of ESAS fully actuate with the exception of AEAS and SRAS. ☑

D Both facilities of ESAS fully actuate the Undervoltage signal only. ☐

Justification

C - CORRECT: When the backup power to VA-30 & VA-40 (VR-11 & 21) momentarily deenergized on the LNP, ESAS saw two channels of safety parameters fail to their accident condition. This meets the 2/4 requirement and will result in a total ESAS actuation.

A - WRONG: Because the RWST level loops get their power from the ESAS sensor cabinets, which never totally lost power, SRAS does not fire.

B - WRONG: All ESAS sensor channels have backup power from the opposite facility. Channel 'A' from VA-10/ 40, Channel 'B' from VA-20/30, Channel 'C' from VA-30/20 and Channel 'D' from VA-40/10.

D - WRONG: The 2/4 requirement and will result in a total ESAS actuation.

References

ESA-01-C, Rev. 3, Pg. 8 & 39 of 73

Comments and Question Modification History

NRC K/A System/E/A

System 013 Engineered Safety Features Actuation System (ESFAS)

Number K2.01 **RO** 3.6* **SRO** 3.8 **CFR Link** (CFR: 41.7)

Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

38 | Origin: Bank | ☑ R | ☐ SRO | ☐ Student Handout? | ☐ Lower Order? | Question ID: 0055661

☑ Selected for Exam ☐ Past NRC Exam?

The plant has tripped from 100% power due to a Large Break - Loss of Coolant Accident.

Which of the following describes the concern if the Containment Spray System is NOT operated per EOP-2532, "Loss Of Coolant Accident"?

A CTMT equipment is not designed for operation in a harsh environment without the CTMT Spray System. ☐

B Containment pressure could exceed the design limit if the CTMT Spray system is not utilized as designed. ☑

C Containment Sump Recirculation will NOT be effective as designed in providing a long term heat sink. ☐

D Containment hydrogen control could jeopardize the CTMT integrity design criteria without spray flow. ☐

Justification

B - CORRECT: 2532 requires that CTMT spray not be secured UNTIL pressure drops to less than 7 psig. It is SECURED to reduce the effects of moisture/water on non-qualified equipment. However, if secured prior to the procedure requirement, CTMT pressure should exceed the design limit.

A - WRONG: All required safety equipment is certified for operation in harsh CTMT environment.

C - WRONG: In a large break LOCA, the long term heat sink will be maintained by SRAS actuation, which aligns RBCCW as the heat sink for HPSI pump/sump recirc.

D - WRONG: Spray flow has minimal effect on CTMT hydrogen concentration or hydrogen burn capability.

References

CSS-00-C, CTMT Spray (Training Mat.), Rev. 4, Ch. 1, Page 5 of 51

Comments and Question Modification History

{Reworded distractor "A" to be more incorrect. - RLC}

NRC K/A System/E/A

System 022 Containment Cooling System (CCS)

Number A1.02 **RO** 3.6 **SRO** 3.8 **CFR Link** (CFR: 41.5 / 45.5)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: Containment pressure

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

38 Origin: ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 0055661**

Org ☐ Selected for Exam ☐ Past NRC Exam?

A Loss of Coolant Accident has occurred, and the Containment Spray System has actuated.||Which of the following is a criteria in the EOP's for securing the Containment Spray System?

- **A** Equipment will be affected by continued operation. ☐
- **B** Containment pressure is less than 7 psig. ☑
- **C** Containment Sump Recirculation initiated. ☐
- **D** Containment hydrogen concentration is greater than 5%. ☐

Justification

2532 requires that ctmt spray be secured at 7 psig (to reduce the effects of moisture/water on non-qualified equipment)$$$$$$

References

Comments and Question Modification History

NRC K/A System/E/A

System Number	RO	SRO	CFR Link

NRC K/A Generic

System Number	RO	SRO	CFR Link

39 | Origin: Bank | ☑ R | ☐ SRO | ☑ Selected for Exam | ☐ Student Handout? | ☐ Past NRC Exam? | ☑ Lower Order? | Question ID: 1000076

During operation in an Event Specific EOP, the Crew encounters a step that states: "IF condition 'X' exists, THEN Go To EOP 2540, Functional Recovery".

Which of the following describes the procedure usage requirements if is determined that condition 'X' does exist?

A Immediately exit the EOP in use, enter EOP 2540, and do not exit until its exit conditions are met. ☑

B Exit the EOP in use, enter EOP 2540 and perform the required actions, then return to the initial EOP. ☐

C Continue in the event based EOP while using EOP 2540 in-parallel to address condition 'X'. ☐

D Complete the section of the EOP in use, then exit that EOP and enter EOP 2540. ☐

Justification

A - CORRECT: OP 2260 states: "The words 'Go To' are used to accomplish branching." Branching is used to direct the user to exit the procedure in use and not return unless otherwise directed.

B - WRONG: 'Go To' does not allow for automatic return to the branching procedure. This must be so directed in the applicable procedure.

C - WRONG: EOP 2540 cannot be used in parallel with any other event based EOP.

D - WRONG: 'Go To' action must be immediately carried out, therefore, the original procedure must be immediately exited.

References

OP-2260, (EOP Users Guide), Rev. 009-001, Page 15

Comments and Question Modification History

NRC K/A System/E/A

System 022 Containment Cooling System (CCS)

Number GS **RO** **SRO** **CFR Link**

SEE GENERIC K/A

NRC K/A Generic

System 2.4 Emergency Procedures /Plan

Number 2.4.5 **RO** 2.9 **SRO** 3.6 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

"Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions."

40

Origin: ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? **Question ID:**
Bank ☑ Selected for Exam ☐ Past NRC Exam? **0056739**

What effect does a SRAS actuation have on the Containment Spray System?

A Containment Spray pumps are tripped and the RBCCW valves to SDC heat exchangers are closed. ☐

B Containment spray pumps are aligned to the containment sump and the RBCCW valves to SDC heat exchangers are closed. ☐

C Containment spray pumps are aligned to the containment sump and the RBCCW valves to SDC heat exchangers are opened. ☑

D Containment spray pumps are tripped and the RBCCW valves to the SDC heat exchangers are opened. ☐

Justification

A. WRONG: CTMT spray are not tripped on a SRAS (the LSPI pumps trip NOT CTMT Spray Pumps) and the RBCCW is supplied to SDC HXs for cooling of recirculated water.

B. WRONG: The first part of the statement is true about the CTMT spray aligned to sump but second part is wrong (same reason as "A" above.

C. CORRECT: Per EOP 2532 step 48, LPSI pumps trip and RBCCW outlet valves from SDC HX open.

D. WRONG: CTMT sprays do NOT trip on SRAS.

References

CSS-00-C, (Training Mat.), Rev. 4, Chg. 1, Page 7 of 51

Comments and Question Modification History

NRC K/A System/E/A

System 026 Containment Spray System (CSS)

Number K1.01 **RO** 4.2 **SRO** 4.2 **CFR Link** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Knowledge of the physical connections and/or cause-effect relationships between the CSS and the following systems: ECCS

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

41 **Origin:** New ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? **Question ID: 6000017**
☑ Selected for Exam ☐ Past NRC Exam?

The plant has tripped from 100% power due to a large break LOCA. The crew is carrying out the actions of EOP-2532, LOCA, and all ESAS equipment has actuated, including SRAS.

Then, the crew notices that HPSI pump performance is degrading due to CTMT sump clogging. The crew wants to stop both CTMT Spray pumps to mitigate the degradation of Safety Injection flow.

In order to secure CTMT Spray during degraded HPSI performance, CTMT pressures must be maintained below which of the following values?

A 1 psig. ☐

B 7 psig. ☐

C 39 psig. ☐

D 54 psig. ☑

Justification

D - CORRECT: Some clogging requires securing of CTMT Spray as soon as possible to limit the degradation of HPSI injection flow.

A - WRONG: This is the TECH SPEC limit for CTMT pressure during normal plant operations.

B - WRONG: This is the normal limit for securing CTMT Spray if sump clogging is not a factor.

C - WRONG: This is the CTMT pressure limit, assuming 54 PSIG is actually 54 PSIA.

References

EOP-2532, Rev. 024, Pg. 42 of 96, Step (contingency) 50.2.a

Comments and Question Modification History

*** Added "degraded HPSI performance" to stem ***
**** Reword stem to imply listed pressures will not be REACHED, as the procedure states them as "less than", not "less than or equal to". ****
{Reworded stem per feedback - RLC}

NRC K/A System/E/A

System 026 Containment Spray System (CSS)

Number K3.02 **RO** 4.2* **SRO** 4.3 **CFR Link** (CFR: 41.7 / 45.6)

Knowledge of the effect that a loss or malfunction of the CSS will have on the following: Recirculation spray system

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

41 | **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? | **Question ID:**
Org ☐ Selected for Exam ☐ Past NRC Exam? | **6000017**

The plant has tripped from 100% power due to a large break LOCA. The crew is carrying out the actions of EOP-2532, LOCA, and all ESAS equipment has actuated, including SRAS.

Then, the crew notices that HPSI pump performance is degrading due to CTMT sump clogging. The crew wants to stop both CTMT Spray pumps to mitigate the degradation of Safety Injection flow.

Which of the following is the MAXIMUM CTMT pressures that is allowed, in order to secure CTMT Spray during degraded HPSI performance?

A 1 psig. ☐

B 7 psig. ☐

C 39 psig. ☐

D 54 psig. ☑

Justification

D - CORRECT: Some clogging requires securing of CTMT Spray as soon as possible to limit the degradation of HPSI injection flow.

A - WRONG: This is the TECH SPEC limit for CTMT pressure during normal plant operations.

B - WRONG: This is the normal limit for securing CTMT Spray if sump clogging is not a factor.

C - WRONG: This is the CTMT pressure limit, assuming 54 PSIG is actually 54 PSIA.

References

EOP-2532, Rev. 024, Pg. 42 of 96, Step (contingency) 50.2.a

Comments and Question Modification History

*** Added "degraded HPSI performance" to stem ***

NRC K/A System/E/A

System 026 Containment Spray System (CSS)

Number K3.02 **RO** 4.2* **SRO** 4.3 **CFR Link** (CFR: 41.7 / 45.6)

Knowledge of the effect that a loss or malfunction of the CSS will have on the following: Recirculation spray system

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

42 | Origin: Mod | ☑ R | ☐ SRO | ☐ Student Handout? | ☐ Lower Order? | Question ID: 6063042
☑ Selected for Exam | ☐ Past NRC Exam?

The plant is operating at 90% power for Control Valve Testing, when a High Pressure turbine 6th stage Extraction Steam supply valve fails closed.

Which of the following is a plant change that would result from this valve failing closed, with NO operator actions?

A Reactor power as seen by the Nuclear Instruments will begin to rise. ☑

B The Main Turbine will soon trip due to loss of heater drain pump flow. ☐

C Actual Main Generator electrical megawatt output will begin to lower. ☐

D Water hammer in the High Pressure Feed Water Heaters will begin. ☐

Justification

A - CORRECT: The extraction steam valve closing will cause feedwater temperature to drop, resulting in positive reactivity addition.

B - WRONG: Heated drain pump flow will be lower, but not enough to trip the main feed pumps on low suction pressure.

C - WRONG: Generator output will rise due to the extraction steam now going through the high pressure turbine.

D - WRONG: Feed water heater water hammer requires loss of all extraction steam.

References

FHD-00-C, (Extraction Steam & Feedwater Heaters Training Mat.), Rev. 4, Page 29 & 30

Comments and Question Modification History

{Modified Stem to ensure CORRECT answer is correct and WRONG choices are wrong. - RLC}

NRC K/A System/E/A

System 039 Main and Reheat Steam System (MRSS)

Number K5.08 **RO** 3.6 **SRO** 3.6 **CFR Link** (CFR: 441.5 / 45.7)

Knowledge of the operational implications of the following concepts as the apply to the MRSS: Effect of steam removal on reactivity

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

42 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Org ☐ Selected for Exam ☐ Past NRC Exam? **6063042**

The plant is operating at 100% power when a High Pressure turbine 3rd stage Extraction Steam supply valve fails closed.

Which of the following is one of the first impacts to the plant of this valve failing closed?

A Reactor power as seen by the Nuclear Instruments will begin to rise. ☑

B The Main Turbine will soon trip on High MSR Drain Tank level. ☐

C Actual Main Generator electrical megawatt output will begin to lower. ☐

D Water hammer in the High Pressure Feed Water Heaters will begin. ☐

Justification

References

Comments and Question Modification History

NRC K/A System/E/A

System 039 Main and Reheat Steam System (MRSS)

Number K5.08 **RO** 3.6 **SRO** 3.6 **CFR Link** (CFR: 441.5 / 45.7)

Knowledge of the operational implications of the following concepts as the apply to the MRSS: Effect of steam removal on reactivity

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

42 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 0063042**

Parent ☐ Selected for Exam ☐ Past NRC Exam?

What is the relationship between the main turbine and the MSR?

A The high pressure turbine exhaust passes through the shell side of the MSR and the second stage extraction steam goes to the first stage of MSR heating. ☑

B The third stage extraction steam passes through the shell side of the MSR and the second stage extraction steam goes to the second stage of MSR heating. ☐

C The high pressure turbine exhaust passes through the shell side of the MSR and the second stage extraction steam goes to the second stage of MSR heating. ☐

D The second stage extraction steam passes through the shell side of the MSR and the third stage of extraction steam goes to the first stage of MSR heating. ☐

Justification

References

Comments and Question Modification History

NRC K/A System/E/A

System Number	RO	SRO	CFR Link

NRC K/A Generic

System Number	RO	SRO	CFR Link

43 | Origin: Bank | ☑ R | ☐ SRO | ☐ Student Handout? | ☐ Lower Order? | Question ID: 0055197
☑ Selected for Exam | ☐ Past NRC Exam?

A plant start up is in progress with power currently at 70%.

The following equipment is in service:

- o 'A' and 'C' Condensate pumps
- o both Heater Drain Tank pumps
- o both Main Feed Pumps.

Then, the flow transmitter for the Condensate Pump Minimum Flow controller fails low.

Without any operator action, what effect does this failure have on the Condensate/Main Feed System?

A Condensate Surge Tank level will RISE. ☐

B SGFP suction pressure will LOWER ☑

C Condenser hotwell level will LOWER ☐

D S/G level will begin to RISE ☐

Justification

B - CORRECT: The transmitter failing low causes the minimum flow control valve to go to its full open position. At least 3200 gpm of condensate pump discharge flow will be diverted to the condenser. This action lowers the suction pressure to the Main Feed pumps.

A - WRONG: This implies confusion between the Condensate Minimum Flow Valve and the Hot Well Reject Valve, which discharges to the Surge Tank. The Condensate Reject Valve taps off of the same pipe extension as the Minimum Flow Valve and is physically located right next to it.

C - WRONG: Condenser level would rise due to the water being diverted back to it, vice going to S/G.

D - WRONG: S/G level would lower due to less flow to it.

References

CON-00-C, Condensate System (Training Mat.), Rev. 7, Ch. 8, Pages 23 & 37

Comments and Question Modification History

**** "A" is arguably correct, based on MFP response to a lowering suction pressure, and must be modified/replaced. ****
{Modified/Replaced choice "A" with a technically wrong distractor. - RLC}

NRC K/A System/E/A

System 056 Condensate System

Number K1.03 **RO** 2.6* **SRO** 2.6 **CFR Link** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Knowledge of the physical connections and/or cause-effect relationships between the Condensate system and the following systems: MFW

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

43 **Origin:** Org ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? ☐ Selected for Exam ☐ Past NRC Exam? **Question ID: 0055197**

A plant start up is in progress with power currently at 70%.

The following equipment is in service:

- o 'A' and 'C' Condensate pumps
- o both Heater Drain Tank pumps
- o both Main Feed Pumps.

Then, the flow transmitter for the Condensate Pump Minimum Flow controller fails low.

Without any operator action, what effect does this failure have on the Condensate/Main Feed System?

A Feedwater Regulating Valve D/P will RISE ☐

B SGFP suction pressure will LOWER ☑

C Condenser hotwell level will LOWER ☐

D S/G level will begin to RISE ☐

Justification

B - CORRECT: The transmitter failing low causes the minimum flow control valve to go to its full open position. At least 3200 gpm of condensate pump discharge flow will be diverted to the condenser. This action lowers the suction pressure to the Main Feed pumps.

A - WRONG: FRV D/P will lower, or possibly be held constant based on SGFP auto speed control.

C - WRONG: Condenser level would rise due to the water being diverted back to it, vice going to S/G.

D - WRONG: S/G level would lower due to less flow to it.

References

CON-00-C, Condensate System (Training Mat.), Rev. 7, Ch. 8, Pages 23 & 37

Comments and Question Modification History

**** "A" is arguably correct, based on MFP response to a lowering suction pressure, and must be modified/replaced. ****

NRC K/A System/E/A

System 056 Condensate System

Number K1.03 **RO** 2.6* **SRO** 2.6 **CFR Link** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Knowledge of the physical connections and/or cause-effect relationships between the Condensate system and the following systems: MFW

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

44 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Bank ☑ Selected for Exam ☐ Past NRC Exam? **1000025**

The following plant conditions exist:

- 80% power increasing at 3% per hour.
- 3 condensate pumps, 5 CPF demins., 2 heater drains pumps, 2 Steam Generator Feed Pumps (SGFP).
- 'A' & 'B' SGFP are in Auto speed control at 4130 and 4050 rpm, respectively.

Then, one of the 'A' SGFP suction pressure detector fails low (0 psi).

Which of the following is a response of the MFW system to this instrument failure, and what action, (if any), will be required to stabilize the plant?

A No immediate response occurs in the MFW system since SGFPs are in speed control. If a second detector were to fail, the 'A' SGFP would trip after five seconds, resulting in a necessary plant trip. ☑

B Both SGFPs shift to Minimum Discharge Pressure control when the detector fails low. The 6th CPF demin. must be placed in service due to higher flow/demin. DP when the 'A' SGFP speeds up. ☐

C 'A' SGFP shifts to Manual and goes to 3400 rpm due to loss of the suction pressure signal. 'A' Main Feed Reg. Valve must be shifted to Manual and adjust to maintain required feed flow condition. ☐

D "A" SGFP will speed up attempting to maintain Feed Reg. Valve DP. However, when the speed exceeds 4200 rpm with the low suction pressure signal, the SGFP will trip requiring a plant trip. ☐

Justification

A - CORRCT: Suction pressure is a critical monitored parameter for the SGFP speed control system. If one is lost, only an alarm is received. However, if a second detector fails, the applicable pump will immediately trip. At 80% power, one SGFP is insufficient to maintain adequate main feed flow necessary to maintain SG level.

B - WRONG: The pumps are in speed control, but suction pressure still has an input to the SGFP control circuit and must be dealt with. Placing a demin in service will have no impact on feed flow as the pump has not changed speed.

C - WRONG: The pump does not shift to manual, however, 3400 rpm is the speed at which auto control takes over.

D - WRONG: SGFPs do not control on main feed suction pressure, however, if 2/3 pressure were to go low, they do have a control input from feed pump speed and would cause the applicable pump to speed up, as stated in the distracter.

References

MFW-00-C, Rev. 8, various locations.
FWL-00-C, Rev. 2, various locations.

Comments and Question Modification History

**** Stem stated "the" suction pressure detector, but answer correctly impied there is MORE than one. ****
{Reworded stem to fix typo, replaced "the" with "one of the". - RLC}

NRC K/A System/E/A

System 059 Main Feedwater (MFW) System

Number A1.07 **RO** 2.5* **SRO** 2.6* **CFR Link** (CFR: 41.5/45.5)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW controls including: Feed Pump speed, including normal control speed for ICS

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

45 | Origin: Mod | ☑ R | ☐ SRO | ☐ Student Handout? | ☐ Lower Order? | Question ID: 6053940
☑ Selected for Exam | ☐ Past NRC Exam?

A plant startup is in progress with current power being held at 1% while preparations are being made to start the first Main Feedwater pump.

Then, VA-10 deenergizes.

Which one of the following describes an effect on the plant at this time, due to this loss of power?

A ONLY #1 SG level will begin to rise without operator action. ☑

B BOTH SG levels will begin to rise without operator action. ☐

C #1 SG AFW Reg. Valve is locked up in it's "throttled" position. ☐

D NEITHER Facility of AFAS can actuate, regardless of SG level. ☐

Justification

A - CORRECT; Both channels of Aux. Feed. Require power to actuate. As Fac. 1 is powered from VA-10, loss of this bus will inop. It. VA-10 also powers the Aux. Fd. Reg. Valve, which will now fail full open, overfeeding the #1 SG.

B - WRONG; Although the loss of VA-10 disables Fac. 1 AFAS, and Fac. 1 AFAS can start the Fac. 2 pump, this facility has no control over the Fac. 2 AFW Reg. Valve specifically. Therefore, there is no change to #2 SG level.

C - WRONG; The #1 Main Feed Reg. Valve locks up on loss of VA-10, not the AFW Reg. Valve. The AFW valve will fail full open on loss of VA-10.

D - WRONG; AFAS Fac. 1 is disabled and Ch. "A" to both facilities is disabled and cannot send an actuation signal. However, Fac. 2 of AFAS can still actuate if two of the remaining three channels ('B', 'C' & 'D') drop below their actuation setpoint.

References

AFW-00-C, Rev. 5, Ch. 3, Pages 11, 18 & 19

Comments and Question Modification History

NRC K/A System/E/A

System 061 Auxiliary / Emergency Feedwater (AFW) System

Number K3.02 **RO** 4.2 **SRO** 4.4 **CFR Link** (CFR: 41.7 / 45.6)

Knowledge of the effect that a loss or malfunction of the AFW will have on the following: S/G

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

45 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Parent ☐ Selected for Exam ☐ Past NRC Exam? **0053940**

The plant is at 100% power, normal operations, when VA-10 is lost.

Assuming all systems respond as expected, which one of the following describe the expected response of the Auxiliary Feedwater System to this loss of power?

A Facility 1 inoperable, Facility 2 in 2/3 logic. ☑

B Facility 1 actuated, Facility 2 in 1/3 logic. ☐

C Facility 1 inoperable, Facility 2 in 1/3 logic. ☐

D Facility 1 in 2/3 logic, Facility 2 is not affected. ☐

Justification

Both channels of Aux. Feed. require power to actuate. As Fac. 1 is powered fron VA-10, loss of this bus will inop it. VA-10 also powers channel "A" sensor input to both facilities, which now require power to actuate. Therefore, Fac. 2 is has one of it's four inputs disabled for a 2/3 logic.

References

Comments and Question Modification History

NRC K/A System/E/A

System Number	RO	SRO	CFR Link

NRC K/A Generic

System Number	RO	SRO	CFR Link

46 | Origin: Bank | ☑ R | ☐ SRO | ☐ Student Handout? | ☐ Lower Order? | Question ID: 0062980

☑ Selected for Exam ☐ Past NRC Exam?

The plant was operating at power when a low Steam Generator water level trip occurred due to a loss of BOTH Main Feedwater Pumps.

In the performance of EOP-2525, the SPO has started both motor-driven AFW pumps and aligned them to feed the SGs. However, upon verifying the AFW flow rate, the SPO notes there is zero (0) feed flow to the either SG.

A Plant Equipment Operator (PEO) is sent to the AFW pumps to monitor, and reports the following:

- Both the "A" and "B" AFW pumps are operating
- Both pumps are extremely noisy
- The pump discharge lines are extremely hot
- Local temperature indicators are reading 265 °F
- NO system valves were found out of position

Which one of the following actions should the Crew initiate?

A With both AFW pumps running, shut the pump manual discharge valves and then slowly open them. ☐

B Override the AFAS, secure the AFW pumps, and attempt to restart each pump on recirculation flow. ☐

C Secure both AFW pumps, shut the AFW pump manual discharge valves and vent the pump casings. ☑

D Shut down either the "A" or "B" AFW pump to provide higher suction pressure to the running AFW pump. ☐

Justification

C - CORRECT; The local indications are that the AFW pumps are steam bound. Per OP-2322, the pumps must be shutdown and vented.

A - WRONG; Local throttling of the AFW pump discharge to more "finely tune" discharge pressure will NOT collapse the steam void inside the pumps.

B - WRONG; Continued operation, in any form, prior to venting has the potential to cause severe damage to the pumps.

D - WRONG; Although the problem is lack of NPSH, this solution will NOT collapse the steam void in the pump casing that is causing the problem.

References

OP-2322, Rev. 026-00, Pg. 50 - 54 of 58

Comments and Question Modification History

NRC K/A System/E/A

System 061 Auxiliary / Emergency Feedwater (AFW) System

Number K6.02 **RO** 2.6 **SRO** 2.7 **CFR Link** (CFR: 41.7 / 45.7)

Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Pumps

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

47 Origin: Bank ☑ R ☐ SRO ☑ Selected for Exam ☐ Student Handout? ☐ Past NRC Exam? ☐ Lower Order? **Question ID: 0073949**

The plant is stable at 98% power following the loss of VA-10 (deenergized). Plant personnel are trouble-shooting the deenergized bus.

Then, the following occurs:

* "A" main steam header breaks in containment and the plant is tripped.
* On the trip, 24C fails to transfer to the RSST due to a failure of the RSST-to-24C Feeder breaker.
* 24C is deenergized.
* ALL other equipment operates as appropriate for the above conditions.
* 2-FW-44 (Aux. Feed Header X-tie) has been closed.
* AFW is aligned to #2 SG using the Turbine Driven Aux. Feed Pump.
* Both electric AFW Pumps are secured.
* #1 Steam Generator (SG) is at 550 psia and dropping.

Then, the STA reports that Main Feed Flow is beginning to show flow to the #1 SG. The US directs the SPO to IMMEDIATELY secure all feed flow to the #1 SG.

Which one of the following actions should the SPO take to immediately secure Main Feed flow to the #1 SG?

A Ensure all Facility 1 SIAS & CIAS actuations on C05. ☐

B Reenergize 4160 VAC buss 24C via 24E and Unit 3. ☐

C Manually isolate #1 Aux. Feed Regulating Valve. ☐

D Ensure all running Condensate Pumps are secured. ☑

Justification

D - CORRECT; The loss of VA-10 locks up the #1 Main Feed Reg. Valve (FRV) in the 100% power position. On the trip, the loss of the RSST to 24C will deenergize 24A and therefore 22A&22C. These 480 load centers power the motor operated valves that are normally used to isolate the locked up #1 FRV. As power to these valves as lost BEFORE any signal could close them, they are all fully open. As soon as the #1 SG depressurizes to </= 500 psia, any running condensate will start feeding it and accelerate the depressurization.

A - WRONG; As power is still lost to these valves, no ESAS or control board signal can effect them. If power WAS available, SIAS & CIAS actuation would already have isolate feed flow.

B - WRONG; The depressurization (and subsequent feeding) will jeopardize CTMT long before any lost bus can be reenergized.

C - WRONG; With the electric Aux feed pumps secured and FW-44 closed, Aux feed is effectively isolated from the #1 SG.

References

AOP-2404C, Rev. 003-06, Pg. 3 of 24. (Main Feed Reg. Valve locks up on loss of VA-10)

Comments and Question Modification History

"A" & "B" not plausible.
[Changed "A" & "B" to improve plausibility. - RLC]

NRC K/A System/E/A

System 062 A.C. Electrical Distribution

Number A1.03 **RO** 2.5 **SRO** 2.8 **CFR Link** (CFR: 41.5 / 45.5)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: Effect on instrumentation and controls of switching power supplies

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

48 Origin: ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? **Question ID:**
New ☑ Selected for Exam ☐ Past NRC Exam? **6000019**

The plant is operating at 100% power with the "B" Emergency Diesel Generator (EDG) running for a Tech. Spec. surveillance. The EDG is in parallel to grid and loaded per the applicable Tech. Spec. surveillance procedure. All EDG parameters are normal and stable.

Then, Unit 3 trips due to a fault on one of their vital AC busses.

Which of the following describes the effect of the Unit 3 trip on the "B" EDG operation?

A The EDG indications must be checked locally to ensure the throttle position and governor are still within required limits. ☐

B The EDG indications must be checked locally to ensure the jacket and oil temperatures are still within required limits. ☐

C The EDG indications on C08 must be checked to ensure the Megawatt loading on the EDG is still within required limits. ☐

D The EDG indications on C08 must be checked to ensure the VAR loading on the EDG is still within required limits. ☑

Justification

D - CORRECT: When Unit 3 trips, it will "dump" all of the VARs is carrying onto the "local" grid, of which the EDG is now attached. The EDG will pick up some of these VARs and have to be adjusted accordingly from C08.

A - WRONG: When the EDG is run for a T.S. surv., it is operated in "Unit Parallel" mode. This means the governor is in a "load set" type of control mode and will maintain EDG MWe loading at a constant setting regardless of how the grid it is tied to fluctuates.

B - WRONG: As the EDG can be assumed to be started and operated per procedure, once the VAR loading is corrected, the EDG should be running fine with all of it's auto systems performing as designed.

C - WRONG: MW loading should not change, regardless of grid instability, with the EDG operating in "Unit Parallel" mode. However, VARS can and will change with the grid fluctuations.

References

SP-2613L, (EDG Tech. Spec. Surveilance), Rev. 00-01, Pg. 13, Note & Caution

Comments and Question Modification History

NRC K/A System/E/A

System 062 A.C. Electrical Distribution

Number A1.01 **RO** 3.4 **SRO** 3.8 **CFR Link** (CFR: 41.5 / 45.5)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: Significance of D/G load limits

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

49

Origin: ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? **Question ID:**
Bank ☑ Selected for Exam ☐ Past NRC Exam? **0053534**

During the performance of EOP 2525, the Secondary Plant Operator noticed that D.C. control power was lost to the "A" Emergency Diesel Generator (EDG).
All local actions were taken in accordance with the procedure to secure the EDG.

When DC control power is restored, which of the following sets of actions must be performed to completely restore the EDG to an operable status?

A Reopen starting air isolation valves and reset the shutdown relay. ☐

B Reset the fuel racks and reset the shutdown relay. ☐

C Reset the fuel racks and reopen the starting air isolation valves. ☐

D Reset the fuel racks, reopen the starting air isolation valves and reset the shutdown relay. ☑

Justification

D - When DC was lost, the EDG fuel racks had to be tripped to stop the engine. A micro switch is activated which sends a signal to energize the SDR when DC is restored. EOP 2525 Contingencies require the DG be tripped locally and the air start valves be closed. In order to restart, the fuel racks have to be reset and the air start valves have to be reopened. Then the SDR is reset by pressing the "alarm reset" button on the skid.

A - WRONG; The fuel racks had to be manually tripped locally on the loss of DC control power in order to shutdown the EDG. Therefore, in order to "restore" the EDG the fuel racks must be manually reset locally. The governor cannot do this automatically.

B - WRONG; The Air start valves must be closed when securing the EDG because the air start solenoids are deenergized. This would cause the EDG to continue to roll in an attempt to start. Therefore, in order to "restore" the EDG, these valves must be reopened.

C - WRONG; The SDR has triggered and must be reset before the EDG governor will allow the EDG to start.

References

EDG-00-C, Emergency Diesel Generators (Training Mat.), Rev. 7, 136 of 143

Comments and Question Modification History

NRC K/A System/E/A

System 063 D.C. Electrical Distribution

Number K2.01 **RO** 2.9* **SRO** 3.1* **CFR Link** (CFR: 41.7)

Knowledge of bus power supplies to the following: Major DC loads

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

50 | Origin: New | ☑ R | ☐ SRO | ☐ Student Handout? | ☑ Lower Order? | Question ID: 6000021
☑ Selected for Exam | ☐ Past NRC Exam?

The plant is in Mode 5 for a refueling outage. DC bus 201A is cross-tied to 201B for replacement of the "B" Battery. ALL other electrical systems are in a normal alignment for a shutdown plant.

Then, the in-service battery charger trips off line on an internal fault.

Which of the following would result from the loss of the battery charger?

A The "A" Battery will be depleted in less than four (4) hours. ☑

B The "A" Battery disconnect fuses will blow due to overload. ☐

C The "A" Battery will carry all loads for about eight (8) hours. ☐

D The inverters will trip on overcurrent in about eight (8) hours ☐

Justification

A - CORRECT: Cross-tying the buses will result in a substantial decrease in battery life. The MODE 5 loads are NOT substantially different than in any other mode.

B - WRONG: The fuses will NOT blow. They will NOT see a high enough current draw when the battery assumes the DC loads, as the battery and fuses are designed to carry both busses for a limited time.

C - WRONG: 8 hours is the normal battery life when carrying only one DC bus, but two are now drawing on the battery.

D - WRONG: The inverters will NOT trip on overcurrent. Although Battery voltage decreases and the power consumed remains the same, current will NOT increase to the point of overload. When the battery voltage drops low enough, the VIAC static switches will shift power sources, taking the load off of the inverters.

References

125 VDC/120 VAC, LVD-00-C, Rev 5, Change 1, Page 12.

Comments and Question Modification History

{New question generated based on lead examiner input. - RLC}
**** Reword "A" to change "deenergized" to "depleted", to utilize terminology more appropriate to a battery. ****
{Replaced "deenergized" with "depleted" in choice "A" - RLC}

NRC K/A System/E/A

System 063 DC Electrical Distribution System

Number A1.01 **RO** 2.5 **SRO** 3.3 **CFR Link** (CFR: 41.5 / 45.5)

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

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

51 | Origin: ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? | Question ID:
Mod ☑ Selected for Exam ☐ Past NRC Exam? 6068511

The "A" EDG is being unloaded and shutdown from the control room following a normal Tech. Spec. surveillance run. However, the EDG is unloaded too quickly and the output breaker trips on reverse power, causing the following annunciator to alarm on Panel C-08:

"DIESEL GEN 12U BKR CLOSING CKT BLOCKED"

No other annunciators are in alarm for the "A" EDG.

Which of the following describes actions that MUST be taken to reset any protective functions for the "A" EDG that this condition may have triggered?

A Reset the "A" EDG output breaker from C08 and verify the associated alarm clears. ☑

B Reset the "A" EDG output breaker from C08, reset the EDG shutdown relay and verify all alarms clear. ☐

C Reset the "A" EDG output breaker trip relays locally at A312 and verify the associated alarm clears. ☐

D Reset the "A" EDG output breaker trip relays locally at A312, reset the EDG shutdown relay and verify all alarms clear ☐

Justification

A - CORRECT: The EDG breaker need only be reset at C08 per the applicable ARP, no further action is necessary to restore the EDG to standby.

B - WRONG: The EDG electronic governor shutdown relay is not actuated by this alarm condition, even though the EDG is rendered inoperable by the triggering of the applicable trip circuit.

C - WRONG: This particular breaker trip will trigger local relays, but they do not have any "interlock" function. They can be reset at any time and do not inop. The EDG in and of themselves.

D - WRONG: This would be the requirement if the reverse power caused a generator fault and actuated the EDG differential-lockout relay, which also disables the EDG.

References

Per ARP 2590F, Rev. 007-07, Page 202 Of 203, CORRECTIVE ACTIONS #2, this the correct method to reset the alarm. Also, classroom training material titled, Emergency Diesel Generator System (EDG-00-C), Revision 7, chg.1 Pages 138 & 139 discusses the closing logic for the diesel output breaker.

Comments and Question Modification History

In Stem, state testing per normal EDG surveillance procedure.
[reworded stem to include suggestion and made question clearer. - RLC]
**** Actions posed in the choices are all plausible based on actual circuit operation. Distinction between right and wrong choice is a matter of detailed "memorization" of Alarm Response Procedure, not a knowledge that distinguishes between a competent and non-competent operator. ****
{Rewrote the question to solicite knowledge of the protective circuit reset requirements, vs. the specific alarm procedure steps. - RLC}

NRC K/A System/E/A

System 064 Emergency Diesel Generators (ED/G)

Number A2.05 **RO** 3.1 **SRO** 3.2* **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loading the ED/G

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

51 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? **Question ID:**
Org ☐ Selected for Exam ☐ Past NRC Exam? **6068511**

The "A" EDG is being unloaded and shutdown from the control room following a normal Tech. Spec. surveillance run. However, the EDG is unloaded too quickly and the output breaker trips on reverse power, causing the following annunciator to alarm on Panel C-08:

"DIESEL GEN 12U BKR CLOSING CKT BLOCKED"

What actions must the SPO take to reset the alarm?

A Ensure "SYN SW, 15G-12U-2 (A312)" in "OFF" and TURN "DG A FDR BKR, 15G-12U-2 (A312)" to "CLOSE" and to "TRIP". ☑

B Ensure "SYN SW, 15G-12U-2 (A312)" in "ON" and TURN "DG A FDR BKR, 15G-12U-2 (A312)" to "CLOSE" and to "TRIP". ☐

C Ensure "SYN SW, 15G-12U-2 (A312)" in "OFF" and TURN "DG A FDR BKR, 15G-12U-2 (A312)" to "TRIP" and to "CLOSE". ☐

D Ensure "SYN SW, 15G-12U-2 (A312)" in "ON" and TURN "DG A FDR BKR, 15G-12U-2 (A312)" to "TRIP" and to "CLOSE". ☐

Justification

A - CORRECT: The correct sequence of switching to meet the reset logic is stated per ARP corrective action step.

B - WRONG: The synch select switch must be in the "OFF" position for the reset to occur.

C - WRONG: Reset will not occur due to the breaker control switch cycled the wrong sequence.

D - WRONG: The synch select switch must be in the "OFF" position for the reset to occur and the reset will not occur due to the breaker control switch cycled the wrong sequence.

References

Per ARP 2590F, Rev. 007-07, Page 202 0f 203, CORRECTIVE ACTIONS #2, this the correct method to reset the alarm. Also, classroom training material titled, Emergency Diesel Generator System (EDG-00-C), Revision 7, chg.1 Pages 138 & 139 discusses the closing logic for the diesel output breaker.

Comments and Question Modification History

In Stem, state testing per normal EDG surveillance procedure.
[reworded stem to include suggestion and made question clearer. - RLC]
**** Actions posed in the choices are all plausible based on actual circuit operation. Distinction between right and wrong choice is a matter of detailed "memorization" of Alarm Response Procedure, not a knowledge that distinguishes between a competent and non-competent operator. ****

NRC K/A System/E/A

System 064 Emergency Diesel Generators (ED/G)

Number A2.05 **RO** 3.1 **SRO** 3.2* **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loading the ED/G

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

51 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Parent ☐ Selected for Exam ☐ Past NRC Exam? **0068511**

The surveillance run on the 'B' Diesel Generator (DG) is being completed. While transferring electrical loads back to the RSST from the DG, the governor is lowered too much and the 'B' DG output breaker trips on reverse power. ||Immediately following the DG output breaker trip, an LNP occurs on bus 24D.|(The timing of these two events is pure chance and does not imply they are connected).||The following annunciators are acknowledged and reported to the US:||·□DIESEL GEN 13U BKR TRIP|·□DIESEL GEN 13U TROUBLE|·□DIESEL GEN 13U BKR CLOSING CKT BLOCKED|||Which one of the following describes the actions required to place the "B" DG in service on bus 24D?||

A No operator actions required other than to ensure the DG automatically starts and loads properly for the LNP condition. | ☐

B Dispatch a PEO to press the alarm reset button on the skid mounted panel and then observe the DG automatically starts and loads properly for the LNP condition. | ☐

C First reset the 'B' DG output breaker using the C08 handswitch, then manually close the breaker to reenergize bus 24D and verify the DG automatically loads properly. | ☑

D Manually stop and then restart the DG from C08, then verify the DG automatically loads properly. | ☐

Justification

The DG will be running but the output bkr tripped on reverse power. This would cause the bkr closing ckt to become blocked. The annunciator response says that if the bkr tripped on reverse power while removing the DG from service, the bkr can be reset. So this would be the proper course of action in this case. If the operator was not allowed to do this, then an Emergency trip due to no SW would have been proper. $$$$ARP 2590F, A30 and A31$$

References

Comments and Question Modification History

NRC K/A System/E/A

System Number	RO	SRO	CFR Link

NRC K/A Generic

System Number	RO	SRO	CFR Link

52 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Mod ☑ Selected for Exam ☐ Past NRC Exam? **6100107**

With the plant operating normally at 100% power, an PROCESS RAD MONITOR HI HI/FAIL alarm is received on C-06/7 . The Radiation Monitor indications on the PPC indicate as follows:

- RM8434B, Aux Building Gaseous Radiation Monitor in alarm and rising.
- RM8999, Radwaste Exhaust Particulate has an Unexplained Rise.
- RM8132B, U2 Stack Gaseous Radiation Monitor has an Unexplained Rise.

The response of these radiation monitors was caused by which of the following?

A The Volume Control Tank vent line is slowly leaking to the Waste Gas header. ☐

B A packing leak on the discharge valve of the running Waste Gas Compressor. ☑

C Evaporation of water in the SFP which lowers level by eighteen inches. ☐

D The inadvertent opening of a relief valve on the running Charging Pump. ☐

Justification

B - CORRECT: Due to the configuration of the Aux Building ventilation system, a packing leak on the discharge valve of the running Waste Gas Compressor would be seen by RM8434B, Aux Building Gaseous Radiation Monitor. The other two Rad Monitors are in the ventilation flow path to the Unit 2 stack, where most Aux Building ventilation is discharged.

A - WRONG: A pinhole leak in the VCT vent to waste gas would result in an alarm on RM8998, Radwaste Exhaust Particulate Rad Monitor, NOT on RM8434B.

C - WRONG: Any activity from the Spent Fuel Pool area would result in an alarm in RM8145A and/or B, Spent Fuel Pool Exhaust Particulate and/or Gaseous Rad Monitor, NOT on RM8434B.

D - WRONG: A radioactive release in the Charging Pump area would result in an alarm on RM8997, Radwaste Exhaust Particulate Rad Monitor, NOT on RM8434B.

References

PPC display printout of Unit Two Rad Monitors.
(Also shown on P&ID 25203-26029, sheet 2, and sheet 3 - not included)

Comments and Question Modification History

*** discuss why "A" is so wrong ***
{Modified 'A' to from "VTC vent line has a small leak" (implies 'to atmosphere' which would make it a correct answer) to "VCT vent line is lsowly leaking" - RLC}

NRC K/A System/E/A

System 073 Process Radiation Monitoring (PRM) System

Number GS **RO** **SRO** **CFR Link**

SEE GENERIC K/A

NRC K/A Generic

System 2.1 Conduct of Operations

Number 2.1.7 **RO** 3.7 **SRO** 4.4 **CFR Link** (CFR: 43.5 / 45.12 / 45.13)

Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation.

52 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 1000107**
Parent ☐ Selected for Exam ☐ Past NRC Exam?

Two days prior to shutdown for a refueling outage, an alarm is received on R8997, Charging Pump Area Ventilation Radiation Monitor. The radiation monitor is reading just above the alarm setpoint.

The following radiation monitors also indicate a relatively small, but steady, unexplained rise:

- R8434B, Aux Building Gaseous Radiation Monitor
- R8998, VCT Area Ventilation Radiation Monitor
- R8132B, U2 Stack Gas Radiation Monitor

The response of these radiation monitors was caused by which of the following:

A A rupture of the waste gas surge tank during RCS degasification. ☐

B A pinhole leak in the degassifier after cooler. ☑

C Evaporation of water in the SFP lowers level by 18 inches. ☐

D Rupture of a piping section during a spent resin transfer to the Spent Resin Tank. ☐

Justification

"A" is incorrect because a rupture of the waste gas surge tank would result in nearly instantaneous alarms on the listed Radiation Monitors, plus others.
"C" is incorrect because a reduction in SFP level will result in the SFP walls drying out. This will cause a gradual rise in particulate activity which would be seen on R8145A, SFP Exhaust Particulate Radiation Monitor and R8132A, U2 Stack Particulate Radiation Monitor.
"D" is incorrect because a rupture of a spent resin line would cause an immediate jump in associated RMs.

References

Comments and Question Modification History

NRC K/A System/E/A

System 060 Accidental Gaseous Radwaste Release

Number AA2.01 **RO** 3.1 **SRO** 3.7 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Accidental Gaseous Radwaste: A radiation-level alarm, as to whether the cause was due to a gradual (in time) signal increase or due to a sudden increase (a spike), including the use of strip-chart recorders, meter and alarm observations

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

53 **Origin:** New ☑ R ☐ SRO ☑ Selected for Exam ☐ Student Handout? ☐ Past NRC Exam? ☐ Lower Order? **Question ID: 6000108**

Which of the following is an accurate list of the Service Water components that receive a SIAS signal and the type of signal received (i.e., start, stop, open closed)?

A
1. ONLY the two in-service Service Water Pumps receive a start signal.
2. Both Diesel Generator Service Water Bypass Valves, 2-SW-231A and B receive an open signal.
3. All three RBCCW TCVs receive an open signal.

☐

B
1. ONLY the two in-service Service Water Pumps receive a start signal.
2. Both Diesel Generator Service Water Supply Valves, 2-SW-89A and B receive an open signal.
3. ONLY the two in-service RBCCW TCVs receive an open signal.

☐

C
1. All three Service Water Pumps receive a start signal.
2. Vital Chiller Crosstie Valves, 2-SW-102 and 104, receive a close signal.
3. ONLY the two in-service RBCCW TCVs receive an open signal.

☐

D
1. All three Service Water Pumps receive a start signal.
2. TBCCW Service Water Supply Valves, 2-SW-3.2A and 3.2B, receive a close signal.
3. All three RBCCW TCVs receive an open signal.

☑

Justification

D - CORRECT: All three SW Pumps receive a start signal to ensure all required emergency heat loads are supplied, even though only the two in-service pumps actually start. The spare pump is prevented from starting by maintaining the control switch in Pull-To-Lock. The Service Water Isolation Valves to TBCCW will close to ensure an adequate supply of Service Water to all components requiring Service water cooling during an accident. All three RBCCW HX TCVs will open to ensure RBCCW has adequate cooling during an accident.

A - WRONG: All three Service Water Pumps receive a start signal, even though only two pumps actually start. Also, the DG Service Water Bypass Valves close on the DG start, NOT from a safety injection actuation signal.

B - WRONG: All three Service Water Pumps receive a start signal, even though only two pumps actually start. Also, the DG Service Water Supply Valves open on the DG start, NOT from a safety injection actuation signal. Additionally, all three RBCCW TCVs get an open signal. The spare HX is outlet isolation is throttle or closed limit or prevent flow through it.

C - WRONG: Vital Chiller Cross tie Valves were removed from the system several years ago. Also, all three RBCCW TCVs get an open signal. The spare HX is outlet isolation is throttle or closed limit or prevent flow through it.

References

SWS-04-C, Service Water System, Rev. 5, Figure 1 (Rev. 1) and Pg. 8, 13, 14 & 15 of 59

Comments and Question Modification History

NRC K/A System/E/A

System 076 Service Water System (SWS)

Number A3.02 **RO** 3.7 **SRO** 3.7 **CFR Link** (CFR: 41.7 / 45.5)

Ability to monitor automatic operation of the SWS, including: Emergency heat loads

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

54 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Bank ☑ Selected for Exam ☐ Past NRC Exam? **1000202**

The Instrument Air supply to the Containment Air Receiver must be isolated to repair an air leak on the supply piping. In order to maintain air to Containment for valve operation, the Station Air System must be aligned to the Containment Air Receiver. A dedicated operator has been selected and briefed.

Cross-tying Station Air to supply the Containment Instrument Air Receiver ______________.

A must be minimized to prevent moisture build up in the Containment Instrument Air System which may impact the operation of air operated valves in Containment ☑

B must be limited to one hour to ensure the CONTAINMENT INTEGRITY Technical Specification action statement is NOT exceeded ☐

C will result in a lower supply pressure to air operated valves in Containment which may cause the valves to operate more slowly during an event ☐

D will require the Containment Instrument Air System to be purged prior to use for breathing air due to the oil vapor contained in the Station Air System ☐

Justification

A - CORRECT: Caution prior to step 4.4.1 in OP 2332A, Station Air, states, "Station Air System has NO air dryer, cross-tying Station Air to supply the Containment Instrument Air Receiver should be minimized to prevent moisture build up in the Containment portion of the Instrument Air System."

B - WRONG: With a dedicated operator stationed, there is NO limit on the CONTAINMENT INTEGRITY Tech Spec LCO.

C - WRONG The Station Air System operates at roughly the same pressure as the Instrument Air System. Additionally, air pressure is regulated at the individual components at a much lower pressure than the system.

D - WRONG: There is NO requirement to purge the Instrument Air System after cross-tying to the Station Air System.

References

OP-2332A, Station Air, Rev. 011-00, Pg. 9 or 13, Caution prior to Step 4.4.1

Comments and Question Modification History

NRC K/A System/E/A

System 078 Instrument Air System (IAS)

Number K1.03 **RO** 3.3* **SRO** 3.4* **CFR Link** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: Containment air

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

55

Origin: New ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? ☑ Selected for Exam ☐ Past NRC Exam?

Question ID: 6000107

The plant is operating normally at 100% power. A small, unisolable Instrument Air leak in Containment has caused Containment air pressure to slowly rise over the past several days. (The Instrument Air leak is NOT large enough to adversely impact the Instrument Air System.)

Which of the following provides the maximum pressure allowed by Technical Specifications and, also, provides the action that must be taken to prevent exceeding this limit?

A 27.7 inches of water. After obtaining Containment samples, vent Containment through the Hydrogen Purge flow path and EBFS to Millstone Stack. ☑

B 27.7 inches of water. After obtaining Containment samples, initiate Containment Purge using EBFS and the Containment Cleanup flow path. ☐

C 20 inches of water. After obtaining Containment samples, vent Containment through the Hydrogen Purge flow path and EBFS to Millstone Stack. ☐

D 20 inches of water. After obtaining Containment samples, initiate Containment Purge using EBFS and the Containment Cleanup flow path. ☐

Justification

A - CORRECT: The Technical Specification limit for Containment pressure is 1.0 psig which equates to 27.7 inches of water. The only approved method of depressurizing Containment in MODE 1 is to use the Hydrogen Purge valves, through EBFAS, out to the Millstone Stack.

B - WRONG: The Containment Cleanup flow path can ONLY be used in MODES 5 or 6.

C - WRONG: 20 inches of water is the maximum pressure referenced in OP 2314B, Containment and Enclosure Building Purge; however, Technical Specifications allow up to 27.7 inches of water (1.0 psig).

D - WRONG: 20 inches of water is the maximum pressure referenced in OP 2314B, Containment and Enclosure Building Purge; however, Technical Specifications allow up to 27.7 inches of water (1.0 psig). Additionally, the Containment Cleanup flow path can ONLY be used in MODES 5 or 6.

References

OP 2314B, Containment and Enclosure Building Purge, Rev. 020-05, Section 4.14, Pg. 39 of 62
Millstone 2 Technical Specification, 3.6.1.4

Comments and Question Modification History

Validate as required RO knowledge. (Make sure not unnecessary knowledge)
[will check - RLC]
{Validators/Reviewers felt knowledge solicited was acceptable. - RLC}

NRC K/A System/E/A

System 103 Containment System

Number A1.01 **RO** 3.7 **SRO** 4.1 **CFR Link** (CFR: 41.5 / 45.5)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system controls including: Containment pressure, temperature, and humidity

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

56 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
New ☑ Selected for Exam ☐ Past NRC Exam? **6000106**

The crew just performed a plant shutdown from 100% power to approximately 3% power in six (6) hours. You have been directed to maintain power between 2% and 4%. After approximately 30 minutes without any operator action, you notice that delta-T power is reading 2.5% and nuclear power is reading 5 E-5%. Remembering your last direction, you withdraw CEAs 10 steps.

How does RCS temperature INITIALLY respond to the change in CEA position and why?

A RCS temperature will rise. Adding positive reactivity at this time will result in a temperature rise because the Reactor is still critical. ☐

B RCS temperature will NOT change. A positive reactivity addition significantly below the Point of Adding Heat will NOT cause a temperature rise. ☑

C RCS temperature will rise. With Delta-T power above the Point of Adding Heat, a positive reactivity addition will cause temperature to rise. ☐

D RCS temperature will NOT change. Plant systems will NOT allow RCS temperature to rise until nuclear power is greater than 5%. ☐

Justification

B - CORRECT: The Reactor is below the point of adding heat, but is still likely critical. Withdrawing CEAs will add positive reactivity, but until the Reactor is above the point of adding heat, RCS temperature will NOT increase.

A - WRONG: Even if the Reactor is critical, RCS temperature will NOT be affected by a positive reactivity addition until power is above the point of adding heat which is around 1% (nuclear) power.

C - WRONG: Nuclear power must be above the point of adding heat for a positive reactivity insertion to cause RCS temperature to rise; NOT Delta-T power.

D - WRONG: RCS temperature will NOT rise until nuclear power is above the point of adding heat (approximately 1%). Plant systems will limit the rise in temperature, but temperature must rise for the systems to respond.

References

Generic Fundamentals (applicable pages supplied to Lead Examiner)

Comments and Question Modification History

NRC K/A System/E/A

System 001 Control Rod Drive System

Number K5.95 **RO** 3.4 **SRO** 3.7 **CFR Link** (CFR: 41.5/45.7)

Knowledge of the following operational implications as they apply to the CRDS: Effect of reactor power changes on RCS temperature

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

57 | Origin: New | ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? ☑ Selected for Exam ☐ Past NRC Exam? | Question ID: 6000105

The plant is in normal operation at 100% power with TWO (2) charging pumps in service in manual (Mode Switch in LVL-1).

Then, the Loop 1Thot input to the Reactor Regulating System suddenly fails low (minimum).

Assuming NO operator action, how will actual Pressurizer level respond and why?

A RISES; Letdown Flow goes to MINIMUM with NO change in Charging Pumps. ☐

B RISES; Letdown Flow goes to MINIMUM and ALL Charging Pumps are running. ☐

C LOWERS; Letdown Flow goes to MAXIMUM with TWO (2) Charging Pump running. ☐

D LOWERS; Letdown Flow goes to MAXIMUM with ONE (1) Charging Pump running. ☑

Justification

When the Thot input to RRS fails low, the RRS Tavg calculator will immediately drop the calculated Tavg based on the "new" RCS temperature input. This will cause the PZR setpoint calculator in RRS to then drop the PZR auto-setpoint output based on the "bad" Tavg. *The auto-setpoint is fed directly to the PZR* Level Controllers and used to control PZR level in normal operation. When the auto-setpoint drops due to the instrument failure, it will cause the in-service PZR Level Controller to "think" actual PZR level is much higher than it should be for the given RCS temperature. The applicable controller will then RAISE Letdown Flow to MAXIMUM (~128 gpm) and STOP any "backup" Charging Pumps that are presently running (40 gpm charging flow). Based on this system response, actual PZR level will LOWER.

D - CORRECT: requires analysis of the failure's effect on the RRS and the effect that the RRS output change would have on the PZR Level Control System.

A - WRONG: implies the malfunction directly effects the Letdown Flow Controller, instead of the PZR Level Controller.

B - WRONG: implies the reverse direction of the input failure, which is a malfunction also covered in Training.

C - WRONG: implies the Charging Pumps are unaffected by the PZR Level Controller when running in LVL-1 mode. The pumps would only be unaffected if they were running for surveillance testing (start on an ESAS signal)

References

RRS-01-C, Reactor Regulating System, Rev. 3, Ch. 3, Page 18

Comments and Question Modification History

NRC K/A System/E/A

System 011 Pressurizer Level Control System (PZR LCS)

Number K6.04 **RO** 3.1 **SRO** 3.1 **CFR Link** (CFR: 41.7 / 45.7)

Knowledge of the effect of a loss or malfunction on the following will have on the PZR LCS: Operation of PZR level controllers

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

58 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? **Question ID:**
Mod ☑ Selected for Exam ☐ Past NRC Exam? **6075300**

Which one of the following CEDS interlocks or design features will be affected by the loss of the Plant Process Computer (PPC)?

A Group Deviation Backup ☐

B Upper Electrical Limit ☐

C Sequential Permissive ☑

D CEA Motion Inhibit ☐

Justification

C - CORRECT: The Sequential Permissive is generated by the PPC to allow withdrawal of the next group of CEAs when its preceding group withdraws above its Upper Sequential Permissive. Without the PPC, this signal is unavailable and CEA groups must be withdrawn in individual mode only.

A - WRONG: Although the PPC generates two different Group Deviation alarms, the CEDS interlock on Group Deviation is generated by CEAPDS (RPIS). Therefore, if the PPC is lost, the alarms are affected but the INTERLOCK is not.

B - WRONG: The Upper Electrical Limit is driven by a separate set of reed switches that also feed the core mimic, NOT the PPC. The Upper Core Stop (sometimes confused with the UEL) is the CEDS interlock that is controlled by the PPC.

D - WRONG: The CMI is a function of CEAPDS.

References

CED-01-C, Rev. 4, Control Element Drive System; Table 5 - Interlocks and Alarms, [or Table 6 - Steps vs. Functions table (end of the document)].

Comments and Question Modification History

NRC K/A System/E/A

System 014 Rod Position Indication System (RPIS)

Number K4.06 **RO** 3.4 **SRO** 3.7 **CFR Link** (CFR: 41.5 / 45.7)

Knowledge of RPIS design feature(s) and/or interlock(s) which provide for the following: Individual and group misalignment

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

58 | Origin: ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? | Question ID: 0075300
Parent ☐ Selected for Exam ☐ Past NRC Exam?

Which one of the following CEDS interlocks will be affected by the loss of the Plant Process Computer (PPC)?

A CEA Motion Inhibit (CMI) ☐

B Upper Electrical Limit (UEL) ☐

C Upper Core Stop (UCS) ☑

D CEA Withdrawal Prohibit (CWP) ☐

Justification

#1; Wrong - The CMI is generated by CEAPDS. Therefore, if the PPC is lost, the CMI is unaffected.
#2; Wrong - UEL is driven by a seperate set of reed switches that also feed the core mimic, not the PPC.
#3; Correct - The Core Stops are driven by pulse counting and controlled by a program running on the PPC. Without the PPC, the interlock is unavailable.
#4; Wrong - the CWP is a function of RPS, driven by 2/4 High Power or TM/LP pretrips.

References

Comments and Question Modification History

NRC K/A System/E/A

System Number	RO	SRO	CFR Link

NRC K/A Generic

System Number	RO	SRO	CFR Link

59 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
New ☑ Selected for Exam ☐ Past NRC Exam? **6000104**

The plant is operating normally at 100% power when 120 Volt Vital AC Bus, VA-20, suddenly deenergizes.

Which of the following describes the affect of a loss of this bus on the Nuclear Instrumentation System?

A Only Channel "B" Wide Range and Channel "B" Power Range Safety Channel are lost. Channel "Y" Power Range Control Channel and Incore Detectors are NOT affected by this loss of power. ☑

B Channel "B" Wide Range, Channel "B" Power Range Safety Channel, and Channel "Y" Power Range Control Channel are lost. Incore Detectors is NOT affected by this loss of power. ☐

C Only Channel "B" Power Range Safety Channel is lost. Channel "B" Wide Range, Channel "Y" Power Range Control Channel, and Incore Detectors are NOT affected by this loss of power. ☐

D Channel "B" Wide Range, Channel "B" Power Range Safety Channel, Channel "Y" Power Range Control Channel, and all the Facility 2 Incore Detectors are lost. ☐

Justification

A - CORRECT: VA-20 supplies Channel "B" Wide Range and Power Range Safety Channel. VR-11 Supplies Channel "Y" Control Channel. The Incore Detectors are self powered.

B - WRONG: Channel "Y" Power Range Control Channel is power from VR-11.

C - WRONG: Channel "B" Wide Range is powered from VA-20

D - WRONG: Channel "Y" Power Range Control Channel is power front VR-11 and there are NO Facility 2 Incore Detectors.

References

NIS-00-C, Rev. 3, Ch. 2, Pg. 8 of 54
AOP-2504D, Loss of 120 VAC Vital Instrument Panel, VA-20, Rev. 003-05, Pg. 21 of 23, Attachment 2.

Comments and Question Modification History

NRC K/A System/E/A

System 015 Nuclear Instrumentation System

Number K2.01 **RO** 3.3 **SRO** 3.7 **CFR Link** (CFR: 41.7)

Knowledge of bus power supplies to the following: NIS channels, components, and interconnections

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

60 Origin: New ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? Question ID: 6000103
☑ Selected for Exam ☐ Past NRC Exam?

A Large Break LOCA concurrent with a loss of the RSST has occurred. EOP 2525 is complete and the crew is performing steps of EOP 2532, Loss of Coolant Accident. Peak Containment pressure was 34 psig, crushing the Containment Air Recirculation Fan ductwork.

Several hours after the event, a Hydrogen concentration of 4.2% is detected in Containment. The RSST is still deenergized.

Which of the following describes the flow path of air to the Hydrogen Recombiners at this point?

A The Containment Auxiliary Recirculation Fans take a suction from the Containment dome discharging to the Containment Air Recirculation Fan suctions. The Containment Air Recirculation Fan discharge ductwork, crushed by the excess Containment pressure, causes the fans to discharge directly to the Hydrogen Recombiners. ☐

B The Post Incident Recirculation Fans are taking a suction on the blowout panels at the discharge of the Containment Air Recirculation Fans. The Post Incident Recirculation Fans discharge directly to the Containment Auxiliary Recirculation Fans which discharge directly to the suction of the Hydrogen Recombiners. ☐

C The Containment Air Recirculation Fans discharge through the open blowout panels to the suction of the Post Incident Recirculation Fans which are discharging to the Containment lower levels to provide mixing. The Hydrogen Recombiners use convection air flow to reduce Containment Hydrogen concentration. ☐

D The Post Incident Recirculation Fans take a suction on the Containment dome and discharge to the Containment Air Recirculation Fans which discharge through the open blowout panels to the lower levels of Containment to provide mixing. The Hydrogen Recombiners use convection air flow to reduce Hydrogen concentration. ☑

Justification

D - CORRECT: The Post Incident Recirculation Fans take a suction on the Containment dome and discharge to the Containment Air Recirculation Fans which discharge through the open blowout panels to the lower levels of Containment to provide mixing. The Hydrogen Recombiners use convection air flow to reduce Hydrogen concentration.

A - WRONG: The Containment Auxiliary Recirculation Fans are deenergized due to the loss of the RSST.

B - WRONG: The Post Incident Recirculation Fans do NOT take a suction form the Containment Air Recirculation Fans, which do NOT discharge directly to the Hydrogen Recombiners.

C - WRONG: The Post Incident Recirculation Fans do NOT take a suction form the Containment Air Recirculation Fans.

References

Containment and Containment Systems, CCS-00-C, Rev. 8, Change 4, Pages 35 & 37. Also Figure 41 (Rev. 1).
Hydrogen Control System, HCS-00-C, Rev. 3, Pages 10 & 11, and Figure 8.2 (Rev. 2)
P&ID 25203-26028, sheet1 and sheet 3 [NOT supplied]

Comments and Question Modification History

NRC K/A System/E/A

System 028 Hydrogen Recombiner and Purge Control System (HRPS)

Number K1.01 **RO** 2.5* **SRO** 2.5 **CFR Link** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Knowledge of the physical connections and/or cause-effect relationships between the HRPS and the following systems: Containment annulus ventilation system (including pressure limits)

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

61 | Origin: New | ☑ R | ☐ SRO | ☐ Student Handout? | ☑ Lower Order? | Question ID: 6000022
☑ Selected for Exam ☐ Past NRC Exam?

The plant has just tripped from 100% power due to a loss of condenser vacuum, with all systems responding as designed.

The US has directed the SPO to stabilize RCS Tavg at 532 °F using the Atmospheric Dump Valves (ADV).

Which of the following is the required automatic setpoint for the ADV controllers to maintain the directed Tavg?

A 900 psia ☑

B 900 psig ☐

C 880 psia ☐

D 920 psia ☐

Justification

A - CORRECT: Per Steam Tables and units on ADV controller, Tavg of 532 °F = 900 psia.

B - WRONG: ADV is upstream of MSIVs, therefore, units are in psia NOT psig like Condenser Dumps.

C - WRONG: Setpoint for Condenser Dumps.

D - WRONG: Normal ADV setpoint, but will maintain Tavg at ~ 534 °F.

References

Steam Tables

Comments and Question Modification History

NRC K/A System/E/A

System 041 Steam Dump System (SDS) and Turbine Bypass Control

Number K5.02 **RO** 2.5 **SRO** 2.8 **CFR Link** (CFR: 41.5 / 45.7)

Knowledge of the operational implications of the following concepts as the apply to the SDS: Use of steam tables for saturation temperature and pressure

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

61 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Org ☐ Selected for Exam ☐ Past NRC Exam? **6000102**

The plant is operating normally at 100%, Middle of Life, when a malfunction in the Steam Dump Tavg Controller, HIC-4165, causes all four Steam Dump Valves to open, resulting in an automatic Reactor trip.

The following parameters were observed:
- Lowest Steam Generator pressure - 553 psia
- Lowest RCS temperature - 478°F
- Lowest RCS pressure - 1685 psia
- Highest Reactor power - 109.8%

Which of the following ensures peak clad temperature will NOT exceed 2200°F during any part of this event?

A A Safety Injection Actuation Signal will be generated at greater than or equal to 572 psia in either Steam Generator. ☐

B In response to the event, the Low Steam Generator Pressure Trip will automatically trip the reactor at greater than or equal to 691 psia. ☐

C The combination of THERMAL POWER, Cold Leg Temperature, and RCS Pressure will NOT exceed the Technical Specification Safety Limit. ☑

D The Axial Shape Index (ASI) was being maintained less than the Local Power Density Technical Specification limit prior to the event. ☐

Justification

C - CORRECT: The Tech Spec Safety Limit is a combination of Thermal Power, Cold Leg Temperature, and Pressurizer Pressure. The Figure for the Reactor Core Thermal Margin Safety Limit is based on preventing overheating of the clad. The maximum power in the figure is NOT exceeded due to the Variable High Power Trip setpoint. The maximum Cold Leg Temperature is NOT exceeded because this event will result in the lowering of RCS temperature. The Plant will remain in the Acceptable Operation region for this event. Remaining in the Acceptable Operation region will prevent overheating of the fuel cladding.

A - WRONG: Although a SIAS will be generated at 572 psia, this actuation is designed to keep the core covered, NOT necessarily to maintain the clad below 2200°F.

B - WRONG: The Low Steam Generator Pressure trip is designed to protect against an excessive rate of heat extraction and subsequent RCS cooldown.

D - WRONG: The Local Power Density limit is used to ensure peak local power density, based on unusual axial power distributions, does not result in fuel centerline melting.

References

Technical Specifications Safety Limit Bases (B 2-1).

Comments and Question Modification History

Reselect K/A due L.E. comments.

NRC K/A System/E/A

System 041 Steam Dump System (SDS) and Turbine Bypass Control

Number K5.06 **RO** 2.5 **SRO** 2.8 **CFR Link** (CFR: 41.5 / 45.7)

Knowledge of the operational implications of the following concepts as the apply to the SDS: Effect of power change on fuel cladding

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

62 | Origin: Mod | ☑ R | ☐ SRO | ☐ Student Handout? | ☑ Lower Order? | Question ID: 6055138
☑ Selected for Exam | ☐ Past NRC Exam?

Shell warming of the main turbine is in progress. Shell pressure is being maintained between 60 and 100 psig.

Which of the following is the correct status of the Turbine Valves?

A The Control Valves are throttled open with the potentiometer, the Main Stop Valves are open, and the Intermediate Stop Valves are open. ☐

B The Control Valves are open, the Main Stop Valves are closed, and #2 Intermediate Stop Valve is throttled open with the potentiometer. ☐

C The Control Valves are open, #2 Main Stop Valve Bypass is throttled open with the potentiometer, and all Intermediate Valves are closed. ☑

D #2 Control Valve Bypass is throttled open with the potentiometer, the Main Stop Valves are closed, and all Intercept Valves are closed. ☐

Justification

C - CORRECT: When the SHELL WARM button is pressed, all 4 Control Valves go fully open. As the potentiometer is rotated, the #2 Main Stop Valve internal bypass opens to admit steam to the Turbine shell. Shell pressure is maintained at 60-100 psig for the duration of Shell Warming All the Intercept and Intercept Stop Valves remain closed to allow the shell to be pressurized to 60-100 psig.

A - WRONG: The Control Valves are NOT throttled open by the potentiometer. Additionally, the Intermediate Stop Valves must remain closed to allow the shell to be pressurized.

B - WRONG: The Main Stop Valves cannot remain closed; otherwise the Shell could NOT be pressurized. Additionally, the potentiometer cannot control any of the Intermediate Valves.

D - WRONG: The potentiometer does NOT control the #2 Control Valve Bypass (No Control Valve Bypass exists). Additionally, the Main Stop Valves cannot remain closed. The shell must be pressurized for Shell Warming.

References

OP 2323A, Main Turbine, Rev. 022-05, Pg. 17 of 61
MTC-00-C, Rev. 5 [NOT included]

Comments and Question Modification History

NRC K/A System/E/A

System 045 Main Turbine Generator (MT/G) System

Number A4.01 **RO** 3.1 **SRO** 2.9 **CFR Link** (CFR: 41.7 / 45.5 to 45.8)

Ability to manually operate and/or monitor in the control room: Turbine valve indicators (throttle, governor, control, stop, intercept), alarms, and annunciators

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

62 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 0055138**
Parent ☐ Selected for Exam ☐ Past NRC Exam?

Shell warming of the main turbine is in progress.||Select the correct description of turbine valve status.||

A Control Valves are closed, Intermediate Stop Valves are open. | ☐

B Control Valves are open, Main Stop Valves are open. | ☐

C Control Valves are open, Intermediate Stop Valves are closed. | ☑

D Control Valves are closed, Intercept Valves are open. | ☐

Justification

Control Valves are open, Intermediate Stop Valves are closed.

References

Comments and Question Modification History

NRC K/A System/E/A

System Number	RO	SRO	CFR Link

NRC K/A Generic

System Number	RO	SRO	CFR Link

63 **Origin: Bank** ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? **Question ID: 0055797**

☑ Selected for Exam ☐ Past NRC Exam?

The SJAE radiation monitor, RE-5099, is located between which of the following components in the Condenser Air Removal system.

A The Mechanical Vacuum pumps (hoggers) and the Condenser Air Removal Fans. ☐

B The SJAEs and the demister. ☑

C The demister and the isolation valve to the Millstone Stack. ☐

D The demister and Fan 55 A & B. ☐

Justification

B - CORRECT: Per the P&ID, this is the order of components in the system.

A - WRONG: MV pumps and Cond. Air Removal Fans are both downstream of the RE-5099.

C - WRONG: Demister and Isolation valves are downstream of the RE-5099.

D - WRONG: Demister and Cond. Air Removal Fans are downstream of the RE-5099 and demister.

References

P&ID for Condenser Air Removal
CAR-00-C, (CAR Training Mat.), Rev. 3, Chg. 3, Figure 1

Comments and Question Modification History

NRC K/A System/E/A

System 055 Condenser Air Removal System (CARS)

Number K1.06 **RO** 2.6 **SRO** 2.6 **CFR Link** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Knowledge of the physical connections and/or cause-effect relationships between the CARS and the following systems: PRM system

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

64 | Origin: New | ☑ R | ☐ SRO | ☐ Student Handout? | ☑ Lower Order? | Question ID: 6000101
☑ Selected for Exam | ☐ Past NRC Exam?

You have been directed to perform a Gaseous Rad Waste discharge of the "A" Waste Gas decay Tank with an OPERABLE Waste Gas Discharge Radiation Monitor (RM-9095).

Which of the following components in the discharge flow path are used to prevent exceeding the pressure limit in the Discharge Radiation Monitor?

A Waste Gas Discharge Isolation Valve, 2-GR-37.2, can be throttled using the variator on Panel C-61. ☐

B Waste Gas Discharge Pressure Controller, 2-GR-9.1 is set to a maximum of 10 psig in the discharge header. ☑

C Waster Gas Discharge Header Stop, 2-GR-71 is manually throttled to maintain 10 psig in the discharge header. ☐

D Radiation Monitor Outlet Isolation, 2-GR-93, is manually throttled to maintain less than 20 psig. ☐

Justification

B - CORRECT: 2-GR-9.1 is the inlet to the discharge header and is procedurally throttled to limit discharge header pressure to 10 psig; thus, limiting the pressure in the Radiation Monitor to 10 psig.

A - WRONG: Although 2-GR-37.2 can be throttled, it is after the Radiation Monitor and will only change the flow rate of the discharge. Procedurally, it is NOT throttled, but fully opened.

C - WRONG: 2-GR-71 is the bypass around the Radiation Monitor and is used to maintain the discharge flow rate below the maximum allowed by the discharge permit.

D - WRONG: 2-GR-93 is used to maintain flow through the Radiation Monitor between 2.5 and 3.5 CFM.

References

SP-2617B, Rev. 011-10, Section 4.1, Pg. 14 of 33.
P&ID 25203-26021, Sh. 2 [NOT provided]

Comments and Question Modification History

Original - K4.03: Knowledge of design feature(s) and/or interlock(s) which provide for the following: Tank loop seals.
Note 5 - Randomly redrawn due to original K/A not fitting system design of MP2.
New K/A - K4.05: Knowledge of design feature(s) and/or interlock(s) which provide for the following: Point of Release

NRC K/A System/E/A

System 071 Waste Gas Disposal System (WGDS)

Number K4.05 **RO** 2.7 **SRO** 3.0 **CFR Link** (CFR: 41.7)

Knowledge of design feature(s) and/or interlock(s) which provide for the following: Point of Release

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

65 **Origin:** New ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? **Question ID:** 6000100
☑ Selected for Exam ☐ Past NRC Exam?

The plant is in normal operation at 100% power, when a Fire System Trouble annunciator is received on C06/7. The ONLY annunciator lit on Fire Panel C-26 is Zone 12, "A" Diesel Room TROUBLE.

Which of the following conditions will cause ONLY this annunciator to alarm?

A The Manual Break Glass station for the "A" Diesel Generator Room has been actuated. ☐

B ALL the Smoke Detectors for the "A' Diesel Generator Room have been actuated. ☐

C ONLY the Heat Detectors for the "A" Diesel Generator Room have been actuated. ☐

D "A" Diesel Room Deluge Valve Isolation Valve, 2-FIRE-22 is closed for surveillance. ☑

Justification

CHOICE (A) - NO
WRONG: Activation of the Manual Break Glass station will cause the Deluge Valve to open, resulting in the actuation of the "A" Diesel Room ALALRM, NOT the TROUBLE annunciator.
VALID DISTRACTOR: An applicant may think that the Manual Break Glass station only results in a TROUBLE annunciator on C-26.

CHOICE (B) - NO
WRONG: Actuation of any 8 smoke detectors in the "A" DG Room will trigger an ALARM annunciator on C-26.
VALID DISTRACTOR: an applicant may think that both heat and smoke detectors are required to actuate an ALARM and a TROUBLE and that only the smoke detectors will trigger a TROUBLE annunciator.

CHOICE (C) - NO
WRONG: Actuation of only the Heat Detectors will trigger an ALARM annunciator on C-26.
VALID DISTRACTOR: an applicant may think that both heat and smoke detectors are required to actuate an ALARM and a TROUBLE and that only the heat detectors will trigger a TROUBLE annunciator.

CHOICE (D) - YES
CORRECT: Closing the "A" Diesel Room Deluge Valve Isolation Valve will actuate a TROUBLE annunciator on C-26.

References

ARP-2590I, "Alarm Response for Fire Panel, C-26" (Zone 12), Revision 02-05, (Pg 20 of 106)

Comments and Question Modification History

NRC K/A System/E/A

System 086 Fire Protection System (FPS)

Number A1.05 **RO** 2.9 **SRO** 3.1 **CFR Link** (CFR: 41.5 / 45.5)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Fire Protection System operating the controls including: FPS Lineups

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

66 **Origin:** **Bank** ☑ **R** ☐ **SRO** ☐ **Student Handout?** ☐ **Lower Order?** ☑ **Selected for Exam** ☑ **Past NRC Exam?** **Question ID: 3100003**

The plant is operating normally at 100% power. A student in training for an RO license is standing a watch under the supervision of the Secondary Plant Operator (SPO).

The Primary Plant Operator (PPO) suddenly becomes ill and has to leave. Due to vacation coverage, a third (work control) Licensed Operator is NOT available. A replacement is called in and will report to working about 30 minutes, however, the PPO is leaving NOW.

Then, five (5) minutes later, the plant trips.

Which of the following statements, as allowed by procedure, describes the Operator responsibilities for this event?

A The SM will perform the duties of the PPO while the US performs the command and control function. ☐

B The US performs the duties of the PPO while the Shift Manager maintains a command and control function. ☑

C The Station Duty Officer performs the command and control function while the US performs the duties of the PPO. ☐

D The SPO trainee will perform the required actions on the primary plant under the direct supervision of the US. ☐

Justification

B - CORRECT: OP 2260, step 1.21.5, states, "In the event it becomes necessary to provide assistance to the board operators, this assistance should be provided by the US in all cases. The Shift Manager should maintain his overall perspective of plant conditions." In this case, a loss of DV-20 will result in a plant trip due to closure of the #2 MSIV. The US should perform the actions of the PPO until relieved.

A - WRONG: The SM should always 'maintain his overall perspective of plant conditions.

C - WRONG: The Station Duty Officer, by definition, is NOT licensed on Unit 2 and CANNOT perform licensed duties.

D - WRONG: The trainee can ONLY 'operate controls with permission of the SM/US and under the direct supervision of a qualified Operator.' The US cannot maintain direct supervision of the trainee during the performance of EOP 2525. Additionally, trainee operation of equipment must be immediately suspended during unanticipated or abnormal events.

References

MP-14-OPS-GDL300
OP 2260, step 1.21.5

Comments and Question Modification History

NRC K/A System/E/A

System 2.1 Conduct of Operations

Number G **RO** **SRO** **CFR Link**

SEE GENERIC K/A

NRC K/A Generic

System 2.1 Conduct of Operations

Number 2.1.2 **RO** 3.0 **SRO** 4.0 **CFR Link** (CFR: 41.10 / 45.13)

Knowledge of operator responsibilities during all modes of plant operation.

67 | Origin: ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? | Question ID:
Bank ☑ Selected for Exam ☐ Past NRC Exam? | 0056587

Shift turnover is in progress and the on-coming Control Operator last had the watch four (4) days ago.

Which of the following items is required to be reviewed by the on-coming Control Operator BEFORE he can assume the watch, per MP-14-OPS-GDL200, Conduct of Operations?

A All Tech. Spec. surveillances performed in the last four (4) days. ☐

B The shift turnover report for the period covering the last 24 hours. ☑

C All Unit Two Condition Reports generated in the last four (4) days. ☐

D Clearances for all tags hung on control boards in the last 24 hours. ☐

Justification

B - CORRECT: MP-14-OPS-GDL200, Conduct of Operations states that the on-coming watch standers must review, as a minimum, the turnover report for the last 24 hours.

A - WRONG: Although the Operations Schedule has Licensed Operators off for four to seven days at a time, review of all T.S. surv. Completed over the last 4 days could take several hours and is totally unnecessary.

C - WRONG: Although the Operations Schedule has Licensed Operators off for four to seven days at a time, review of all CRs entered over the last 4 days could take several hours and is totally unnecessary.

D - WRONG: The number of tags hung on the boards could number close to a hundred, too much time would be required for review and is not necessary.

References

MP-14-OPS-GDL200, Rev. 012-01, Conduct of Operations, Pg. 15

Comments and Question Modification History

NRC K/A System/E/A

System 2.1 Conduct of Operations

Number G **RO** **SRO** **CFR Link**

SEE GENERIC K/A

NRC K/A Generic

System 2.1 Conduct of Operations

Number 2.1.3 **RO** 3.0 **SRO** 3.4 **CFR Link** (CFR: 41.10 / 45.13)

Knowledge of shift turnover practices.

68 | Origin: Mod | ☑ R | ☐ SRO | ☐ Student Handout? | ☐ Lower Order? | Question ID: 6100074
☑ Selected for Exam | ☐ Past NRC Exam?

The plant has tripped due to an Excess Steam Demand Event in CTMT.
On the trip, off-site power was lost and the "A" EDG output breaker would NOT close. The SPO then emergency tripped the 'A' EDG.
All other equipment responded as designed.
The crew completed EOP-2525, diagnosed an ESD and entered EOP-2536.

Twenty (20) minutes after the plant trip, the following conditions now exist:

- The "A" EDG output breaker has been repaired and is cleared for operation.
- The SPO has been directed to start and load the "A" EDG and then restore Facility 1 equipment.
- CTMT pressure is 23 psig and slowly dropping.

Which of the following is correct, with respect to restoring Facility 1 equipment?

A Start the "A" EDG and verify the sequencer energizes bus 24C, starts the 'A' SW pump and then the "A" RBCCW pump. ☐

B Close the RBCCW supply to "A" SDC Heat Exchanger, 2-RB-13.1A., BEFORE allowing the "A" RBCCW pump to start. ☐

C After restoring 24C and Fac. 1 SW, start and run the Fac. 1 CAR fans for >15 minutes prior to starting Facility 1 RBCCW. ☐

D Start the "A" EDG and energize 24C; restore Facility 1 SW flow, but do NOT restore Fac. 1 RBCCW to the CAR fans. ☑

Justification

D - CORRECT: CTMT pressure is >20#. The reason for not starting the RBCCW pump is water hammer caused by the hot RBCCW in idle CAR cooler flashing on pp start.

A - WRONG: This will start the RBCCW pump before the RBCCW in the CAR fan heat exchanger can be cooled.

B - WRONG: The SDC has no RBCCW flow at this time and 2-RB-13.1 is not open. Even so, the RBCCW in this heat exchanger is not the problem.

C - WRONG: Running the CAR fan will NOT cool the RBCCW in it's heat exchanger enough if CTMT pressure is still above 20 psig.

References

EOP-2525, Rev. 20, Pg. 5 of 26, Contingency Action

Comments and Question Modification History

NRC K/A System/E/A

System 2.1 Conduct of Operations
Number G **RO** **SRO** **CFR Link**
SEE GENERIC K/A

NRC K/A Generic

System 2.1 Conduct of Operations

Number 2.1.20 **RO** 4.3 **SRO** 4.2 **CFR Link** (CFR: 41.10 / 43.5 / 45.12)
Ability to execute procedure steps.

68 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID: 1000074**

Parent ☐ Selected for Exam ☐ Past NRC Exam?

The plant experienced a small break LOCA concurrent with an ESDE.
On the trip, off-site power was lost and the 'A' EDG output breaker failed to close.
All Facility 2 equipment responded as designed.
The crew completed 2525, diagnosed 2 events and entered EOP 2540.
Two of the 'B' header HPSI valves have been throttled shut and the 'B' CS pump has been secured.
After the 'A' EDG output breaker was repaired the SPO is restoring vital auxiliaries using EOP 2540B, MVA-AC-3.
The EDG was started, closed onto bus 24C, and the 'A' SW pump has been started.

With respect to Facility 1 RBCCW, what is the correct action to be performed?

A Throttle the 'A' RBCCW pump discharge to 10%, start the pump, and throttle the discharge open. ☑

B Close RBCCW Supply to "A" Shutdown Cooling Heat Exchanger, 2-RB-13.1A. ☐

C Start and run both Facility 1 Containment Air Recirculation fans >15 minutes prior to starting Facility 1 RBCCW. ☐

D Check RBCCW surge tank level >40% then start Facility 1 RBCCW pump at full flow. ☐

Justification

A: correct, CTMT pressure is <20# (CS PP off), pp has been off >5 mins requires throttling discharge on start; B: valve will NOT receive an open signal on the SIAS; therefore it does NOT need to be closed. Student may believe it needs to be closed to limit RBCCW flow through the Facility 1 header.; C: area of concern for water hammer is hot RBCCW in idle CAR cooler flashing on pp start, running fans would cool the water; D: ensuring surge tank level >40% addresses subcooled margin for heated water pocket in CAR coolers

References

Comments and Question Modification History

NRC K/A System/E/A

System Number	RO	SRO	CFR Link

NRC K/A Generic

System Number	RO	SRO	CFR Link

69 Origin: Bank ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? ☑ Selected for Exam ☐ Past NRC Exam? Question ID: 0053322

The PPO is about to perform SP-2601D, RPS - Calorimetric calibration.

Which of the following groups contains ONLY those parameters that are directly affected by adjustments made to the NI or Delta-T power calibration potentiometers? (Assume that only the calibration adjustments are made, with no other operator actions.)

A CEAPDS PDIL indication and Tave/Tref recorder Power indication ☐

B Power Ratio Calculator recorder ASI setpoint and Calorimetric Power indication ☐

C LPD trip setpoint and CEA Motion Inhibit on PDIL "trigger" setpoint ☑

D Linear Range Power indication and Diverse Scram Aux. Feed actuation setpoint ☐

Justification

C - CORRECT: NI or Delta-T potentiometers will affect Q-power. Q-power is not an input to the Tavg/Tref recorder or the wide range power drawers.

A - WRONG: PDIL setpoint is fed by RPS, not the actual indication. Also, power indication on the recorders is delta-T power.

B - WRONG: The PRR ASI setpoint is fed by the highest of the Q-Powers from all four channels, however, the Calorimetric calculation of the PPC does not receive and input from RPS.

D - WRONG: Linear power indication is directly off the NI's, therefore, unaffected by the Q-power Cal. Pot.

References

RPS-01-C, Reactor Protection System (Training Mat.), Rev. 6, Ch. 2, Page 52 of 80

Comments and Question Modification History

NRC K/A System/E/A

System 2.2 Equipment Control

Number G **RO** **SRO** **CFR Link**

SEE GENERIC K/A

NRC K/A Generic

System 2.2 Equipment Control

Number 2.2.12 **RO** 3.0 **SRO** 3.4 **CFR Link** (CFR: 41.10 / 45.13)

Knowledge of surveillance procedures.

70 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? **Question ID:**
Bank ☑ Selected for Exam ☐ Past NRC Exam? **0054308**

The 'A' RBCCW pump motor is being red tagged out of service for maintenance during which the supply breaker will be completely removed from its cubicle for repair. The Work Order does NOT allow for testing at this time.

Which of the following is the proper method for hanging the tags for the 'A' RBCCW pump motor?

A Hang a BLUE tag on the control board switch and a RED tag at the breaker cubicle. ☐

B Hang a RED tag on the control board switch and a YELLOW tag at the breaker cubicle. ☐

C Hang a RED tag on the control board switch and a BLUE tag at the breaker cubicle. ☐

D Hang a YELLOW tag on the control board switch and a RED tag at the breaker cubicle. ☑

Justification

D - CORRECT: WC-2 requires that personnel "hang tags for removed breakers on [the} breaker cubicle." Yellow tags are used on the hand switches in the Control Room to signify a component tagged out in the plant.

A - WRONG: Ordinarily, if testing were allowed, a Blue tag would be hung on the control board to allow for this. However, testing is not allow per the stem.

B - WRONG: If the Switch were also part of the maintenance and troubleshooting package then this would be correct. However, that has to be specifically stated.

C - WRONG: This configuration could be used if the breaker were to be locally tested by Maintenance.

References

WC-2,Rev. 006-03, Step 1.6.4

Comments and Question Modification History

NRC K/A System/E/A

System 2.2 Equipment Control

Number G **RO** **SRO** **CFR Link**

SEE GENERIC K/A

NRC K/A Generic

System 2.2 Equipment Control

Number 2.2.13 **RO** 3.6 **SRO** 3.8 **CFR Link** (CFR: 41.10 / 45.13)

Knowledge of tagging and clearance procedures.

71 | Origin: Bank | ☑ R | ☐ SRO | ☐ Student Handout? | ☑ Lower Order? | Question ID: 0054856
☑ Selected for Exam | ☐ Past NRC Exam?

A Fuel Handling Accident has occurred inside the Unit #2 Containment Building. All personnel were immediately evacuated.
The containment isolation was completed effectively. Management personnel are interested in making a Containment Entry to evaluate the condition and location of the dropped fuel assembly.

Which radiation monitor(s) should be utilized in evaluating conditions inside the Containment, prior to the actual entry?

A Unit 2 Stack Radiation Monitors ☐

B CTMT High Range Radiation Monitors ☐

C Millstone Stack Radiation Monitor ☐

D Refueling Platform Radiation Monitor ☑

Justification

D - CORRECT: Refuel platform is an area rad. Mon. that has a range capable of detecting CTMT rads. From fuel leaking.

A - WRONG: The stack would not see the rads. Because CTMT would not be vented through normal ventilation (Main Exhaust) while containing fission product gasses.

B - WRONG: The High Range monitors have too high a range for just fission product gasses. They are designed to see levels in the hundreds of rads.

C - WRONG: Millstone stack would see the rads., but the stack is used by both units and is substantially diluted by normal discharge.

References

AOP-2577 (Fuel Handling Accident), Rev. 008-01, Section 2.0

Comments and Question Modification History

[Reworded "B" & "C" per L.E. comments - RLC]
*** The "Service" Platform Radiation Monitor is commonly refered to as the "Refueling" Platform Radiation Monitor by operations. ***
{Changed "Service" to "Refueling" on "D". - RLC}

NRC K/A System/E/A

System 2.2 Equipment Control

Number G **RO** **SRO** **CFR Link**

SEE GENERIC K/A

NRC K/A Generic

System 2.2 Equipment Control

Number 2.2.30 **RO** 3.5 **SRO** 3.3 **CFR Link** (CFR: 45.12)

"Knowledge of RO duties in the control room during fuel handling such as alarms from fuel handling area, communication with fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation."

72 **Origin:** ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Bank ☑ Selected for Exam ☐ Past NRC Exam? **2000021**

A minor fuel handling accident has just occurred in the Spent Fuel Pool and gaseous activity is bubbling up to the surface.

The unit supervisor directs the PPO to verify equipment actuation preventing/mitigating the discharge of activity to the environment.

Which of the following system alignments must the PPO verify to accomplish this task?

A Fan F-20 supplying outside air, discharge aligned to the EBFAS. ☐

B Fan F-20 off with the outside air supply isolated, discharge aligned to EBFAS. ☑

C Fan F-20 supplying outside air, discharge aligned to the Main Exhaust System. ☐

D Fan F-20 off with the outside air supply isolated, discharge aligned to the Main Exhaust System. ☐

Justification

B - CORRECT: 2 of 4 SFP area RMs exceeding 50 mR/hr actuates AEAS which places the SFP area under a negative pressure by securing supply air (F-20 off and outside air dampers closed) and aligning suction to EBFS so any activity will be filtered and monitored.

A - WRONG: F-20 must be secured because it could pressurize the area around the SFP and cause air leakage directly to the environment.

C - WRONG: See "A" and "D", both concepts are wrong.

D - WRONG: The discharge would not be filtered enough (no activated charcoal for the iodine) if sent to Main Exhaust. Also, it would be a "ground" release.

References

AOP-2577 (Fuel Handling Accident), Rev. 008-01, Attachment 2, Pg. 10

Comments and Question Modification History

NRC K/A System/E/A

System 2.3 Radiation Control
Number G **RO** **SRO** **CFR Link**
SEE GENERIC K/A

NRC K/A Generic

System 2.3 Radiation Control

Number 2.3.11 **RO** 2.7 **SRO** 3.2 **CFR Link** (CFR: 45.9 / 45.10)
Ability to control radiation releases.

73 **Origin:** Mod ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:** 6500023
☑ Selected for Exam ☐ Past NRC Exam?

Preparations for a plant shutdown from 100% is in progress due to a Tech. Spec. LCO requirement. The IDENTIFIED leakage from the RCS to the Primary Drain Tank is in excess of the RCS Leakage Tech. Spec. The US has directed the PPO to start forcing Pressurizer sprays, start a second charging pump in manual, and notify HP (per the guidance in AOP-2575, Rapid Down Power).

Which of the following describes the REASON for the notification of the Health Physics Department, per AOP-2575?

A The expected background radiation rise due to the rise in the RCS identified leakage. ☐

B The expected lowering of radiation levels in the plant due to the upcoming shutdown. ☐

C The expected radiation rise when balancing letdown flow for the second charging pump. ☑

D The expected radiation rise from dumping RCS to radwaste as PZR level setpoint lowers. ☐

Justification

The "Parent question (# 5000023) was SRO ONLY, this Modification is not.

C - CORRECT: Starting a 2nd pump requires increasing letdown flow. The higher flow rate would cause a rise in rad. Levels due to the higher amount of N-16.

A - WRONG: Leakage to the PDT would not be seen outside CTMT until the PDT is pumped to rad waste.

B - WRONG: HP will be informed of the rad. Level changes of the shutdown when the shutdown is discussed (brief). This will be done in several minutes by the US.

D - WRONG: PZR setpoint does not lower as fast as the PZR level lowers due to the RCS cooldown on the power reduction. Therefore, there will be no radwaste generated by the shutdown.

References

1. RPM-1.1.2, "Radiation Protection Program and ALARA Program", Revision 3 (8/19/04) (Pg 5,8,9,10,11,16 of 33)
2. OP-2304E, "Charging Pumps", Revision 15 (03/09/04), Step 4.4.10 (Pg 21 of 25)

Comments and Question Modification History

[Modified "C" & "D" based on Lead Examiner feedback - RLC]

NRC K/A System/E/A

System 2.3 Radiation Control

Number G **RO** **SRO** **CFR Link**

SEE GENERIC K/A

NRC K/A Generic

System 2.3 Radiation Control

Number 2.3.10 **RO** 2.9 **SRO** 3.3 **CFR Link** (CFR: 43.4 / 45.10)

Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.

73 | Origin: ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? | Question ID:
Parent ☐ Selected for Exam ☐ Past NRC Exam? | 5000023

Which of the following evolutions raises an immediate ALARA concern requiring notification of Health Physics Department? Consider the effects of the described action only.

A increasing CVCS letdown flow during normal power operations ☑

B starting of the HPSI pumps by SIAS during a large break LOCA ☐

C increasing SFP cooling flow during spent fuel pool fuel moves ☐

D shifting from 'C' to 'A' Charging Pump running at 75% power ☐

Justification

SRO ONLY QUESTION - Samples 55.43(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

CHOICE (A) - YES
Step 4.4.10 requires notification of Health Physics Department of any change in charging or letdown flow.

CHOICE (B) - NO
WRONG: HPSI pumps are aligned for injection from the RWST and will not affect local dose rates until post-SRAS. EOP does not require HP notification at start of LOCA
VALID DISTRACTOR: A LOCA has the potential to raise local dose rates.

CHOICE (C) - NO
'WRONG: Notification of HP is not required for increasing SFP cooling flow.
VALID DISTRACTOR: Plausible that increasing SFP cooling flow might create ALARA concerns.

CHOICE (D) - NO
WRONG: Shifting charging pumps does not change charging flowrate and therefore does not present ALARA concerns. Both of these pumps are located in the same general area.
VALID DISTRACTOR: A caution states that HP should be notified for changing charging flow conditions.

References

1. RPM-1.1.2, "Radiation Protection Program and ALARA Program", Revision 3 (8/19/04) (Pg 5,8,9,10,11,16 of 33)
2. OP-2304E, "Charging Pumps", Revision 15 (03/09/04), Step 4.4.10 (Pg 21 of 25)

Comments and Question Modification History

NRC K/A System/E/A

System 2.3 Radiation Control
Number G **RO** **SRO** **CFR Link**
SEE GENERIC K/A

NRC K/A Generic

System 2.3 Radiation Control
Number 2.3.2 **RO** 2.5 **SRO** 2.9 **CFR Link** (CFR: 41.12 / 43.4. 45.9 / 45.10)
Knowledge of facility ALARA program.

74 Origin: Bank ☑ R ☐ SRO ☐ Student Handout? ☑ Lower Order? ☑ Selected for Exam ☐ Past NRC Exam? **Question ID: 0071925**

Following a trip from full power due to loss of all feedwater, stopping RCPs, a major step in the mitigation strategy of EOP 2537, precedes the Contingency Action to initiate once through cooling.

Which of the following describes the basis for stopping RCPs at this point in the procedure?

A To prevent RCP operation outside of design limits during the upcoming RCS depressurization. ☐

B To minimize RCP seal damage due to the impending CIAS which will isolate seal bleedoff flow. ☐

C To eliminate RCP heat input into the RCS and to conserve the remaining SG water inventory. ☑

D To minimize subsequent RCS voiding and mass loss if Once Through Cooling must be initiated. ☐

Justification

C - CORRECT: RCS heat input is the primary bases for the securing of ALL RCPs immediately after implementing EOP-2537.

A - WRONG: If depressurization were a concern, the RCPs would be secured at the same points in the depressurization as is done in a normal plant cooldown.

B - WRONG: Once CIAS has initiated and isolated the normal RCP Bleedoff flow path to the VCT, RCP seal bleedoff flow can be sent to the PDT via the CTMT bleedoff flow relief valve.

D - WRONG: The potential for accelerated mass loss of a Small Break LOCA (TMI scenario) is real, but RCPs are required to be secured long before the initiation of OTC and RCS pressure reaches the SIAS initiation setpoint.

References

E37-01-C, Loss Of All Feedwater (Training Mat.), Rev. 1, Ch. 2, Page 13 of 28

Comments and Question Modification History

NRC K/A System/E/A

System 2.4 Emergency Procedure /Plan

Number G **RO** **SRO** **CFR Link**

SEE GENERIC K/A

NRC K/A Generic

System 2.4 Emergency Procedures /Plan

Number 2.4.23 **RO** 2.8 **SRO** 3.8 **CFR Link** (CFR: 41.10 / 45.13)

Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.

75 | Origin: Bank | ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? ☑ Selected for Exam ☑ Past NRC Exam? | Question ID: 3100007

The plant is operating normally at 100% power. I&C personnel have removed the #1 'alternate' steam flow detector for a scheduled calibration. The #1 steam flow selector switch was placed in the 'MAIN' position on C-05 prior to the start of the calibration.

Approximately 15 minutes after the start of the calibration, the SPO reports the following:
- SG LEVEL SET POINT DEVIATION HI/LO annunciator
- #1 and #2 SG levels are lowering.
- #1 SG main steam flow indication on the SGTR screen is lowering and appears to be caused by a failing detector.

Which of the following actions must the SPO take to stabilize the plant?

A Take manual control of #1 FRV. Place BOTH SGFP speed controls in 'MANUAL' and maintain FRV D/P to between 20 and 100 psid while manually restoring #1 SG level. ☑

B Manually control #1 SG FRV and raise the setpoint of #2 FRV, to restore SG levels to normal level. Ensure the SGFPs automatically maintain the required delta-P. ☐

C Take manual control of the #1 FRV and the "A" SGFP. Restore #1 SG level to 70% while operating the "A" SGFP in manual to maintain the required delta-P. ☐

D Take manual control of BOTH SG FRVs and position them to restore SG levels. Open the CPF bypass valve to raise SGFP suction pressure and compensate for the greater demand in feed flow. ☐

Justification

A - CORRECT: The steam flow detector is an input to automatic control of SGFP speed as well as automatic FRV position. ARP 2590D requires manual control of SGFP speeds and manual control of BOTH FRVs, until SG levels are stabilized, on a failure of the steam flow detector.

B - WRONG: There is nothing directly wrong with the #2 SG FRV control system. However, the # 1 FRV changing position due to the failure, along with the SGFPs changing speed, requires placing the #2 FRV in manual a necessary burden to the recovery of the plant. NOT placing both SGFPs in manual will severely jeopardize the mitigation of this event.

C - WRONG: BOTH SGFPs need to be manually controlled to mitigate this failure. Controlling only one will allow the other to shut down, and one SGFP cannot supply sufficient feed flow above 75% power.

D - WRONG: Controlling #2 SG FRV in manual is counter-productive. Opening the CPF bypass will raise SGFP suction pressure, but the control system will then LOWER SGFP speed even faster in an attempt to maintain a perceived lower need of delta-P.

References

ARP-2590D (C05 Alarm Response), Rev. 002-12, D-16, Pg. 064
OP-2385 (Feedwater Control System Operation), Rev. 010-00, Page

Comments and Question Modification History

*** Controlling both FRV to to restore SG levels is an arguably correct answer ***
{Modified all four choices to conform with latest procedure revision - RLC}
**** Failing steam flow detector must be the "main" (though NOT stated) of there is no correct answer, which is to monitor and do nothing. ****
{Added the word "main" to stem to more clearly designate the applicable failing detector - RLC}

NRC K/A System/E/A

System 2.4 Emergency Procedure /Plan

Number G **RO** **SRO** **CFR Link**

SEE GENERIC K/A

NRC K/A Generic

System 2.4 Emergency Procedures /Plan

Number 2.4.50 **RO** 3.3 **SRO** 3.3 **CFR Link** (CFR: 45.3)

Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

75

Origin: ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Bank ☑ Selected for Exam ☑ Past NRC Exam? **3100007**

75

Origin: ☑ R ☐ SRO ☐ Student Handout? ☐ Lower Order? **Question ID:**
Org ☐ Selected for Exam ☐ Past NRC Exam? **3100007**

The plant is operating normally at 100% power. I&C personnel have removed the #1 'alternate' steam flow detector for a scheduled calibration. The #1 steam flow selector switch was placed in the 'MAIN' position on C-05 prior to the start of the calibration.

Approximately 15 minutes after the start of the calibration, the SPO reports the following:
- SG LEVEL SET POINT DEVEATION HI/LO annunciator
- #1 and #2 SG levels are lowering.
- #1 SG steam flow indication on the SGTR screen is lowering and appears to be caused by a failing detector.

Which of the following actions must the SPO take to stabilize the plant?

A Take manual control of #1 FRV. Place BOTH SGFP speed controls in 'MANUAL' and maintain FRV D/P to between 20 and 100 psid while manually restoring #1 SG level to 70%. ☑

B Take 'MANUAL' control of BOTH SG FRVs and position them to restore SG levels to the normal level band. Ensure the SGFPs automatically maintain the required delta-P. ☐

C Take 'MANUAL' control of the #1 FRV and the "A" SGFP. Restore #1 SG level to 70% while operating the "A" SGFP in manual to maintain the required delta-P. ☐

D Take 'MANUAL' control of BOTH SG FRVs and position them to restore SG levels. Open the CPF bypass valve to raise SGFP suction and discharge pressure and compensate for the greater demand in feed flow. ☐

Justification

A - CORRECT: The steam flow detector is an input to automatic control of SGFP speed as well as automatic FRV position. ARP 2590D requires manual control of SGFP speeds and manual control of #1 FRV on a failure of the steam flow detector.

B - WRONG: There is nothing wrong with the #2 SG FRV control system, placing it in manual adds an unnecessary burden to the recovery of the plant. NOT placing both SGFPs in manual will prevent operator mitigation of this event.

C - WRONG: BOTH SGFPs need to be manually controlled to mitigate this failure. Controlling only one will allow the other to shut down, and one SGFP cannot supply sufficient feed flow above 75% power.

D - WRONG: Controlling #2 SG FRV in manual is counter-productive. Opening the CPF bypass will raise SGFP suction pressure, but the control system will then LOWER SGFP speed even faster in an attempt to maintain a perceived lower need of delta-P.

References

ARP-2590D (C05 Alarm Response), Rev. 002-12, D-16, Pg. 064
OP-2385 (Feedwater Control System Operation), Rev. 010-00, Page

Comments and Question Modification History

*** Change "A" to say "both FRV to MANUAL ***

NRC K/A System/E/A

System 2.4 Emergency Procedure /Plan
Number G **RO** **SRO** **CFR Link**
SEE GENERIC K/A

NRC K/A Generic

System 2.4 Emergency Procedures /Plan

Number 2.4.50 **RO** 3.3 **SRO** 3.3 **CFR Link** (CFR: 45.3)
Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.