Question #: 1 | Question ID: 65167 | ☑ RO ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 4 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

Unit 2 was operating at 100% power when an electrical transient occurred. Given the following conditions and events in sequence:

- VA-20 was deenergized
- The plant tripped
- MSI actuated
- SGTR occurred on the #1 Steam Generator (SG)
- Upon reaching step 6 of EOP 2525 (SPTA) the BOP was directed to feed the #2 SG using Aux Feed Water (AFW)

Which one of the following statements correctly describes:
1. the required actions, and
2. the correct procedure to be used.

☐ **A** 1. Actions: Place both AFW "OVERRIDE/MAN/START/ RESET" hand switches in "Pull-To-Lock", then close Aux Feed Header Crosstie, 2-FW-44, and feed #2 SG with the turbine driven AFW pump only.
2. Procedure: EOP-2541, Appendix 6 (TDAFW Pump Normal Startup).

☐ **B** 1. Actions: Manually initiate all Facility 2 AFW components, then close Aux Feed Header Crosstie, 2-FW-44, and feed #2 SG with the turbine driven AFW pump only.
2. Procedure: EOP-2541, Appendix 7 (TDAFW Pump Abnormal Startup).

☑ **C** 1. Actions: Manually start both MDAFW pumps, place both AFW "OVERRIDE/MAN/START/ RESET" hand switches in "Pull-To-Lock", then control #1 AFW Regulating Valve in manual and have the #2 AFW Regulating Valve controlled locally.
2. Procedure: EOP-2525, step 6, (without starting the TDAFW pump).

☐ **D** 1. Actions: Place Facility 2 AFW "OVERRIDE/MAN/START/ RESET" hand switch in "Pull-To-Lock", then control #1 AFW Regulating Valve in manual and have the #2 AFW Regulating Valve controlled locally, feeding with the turbine driven AFW pump only.
2. Procedure: EOP-2541, Appendix 7 (TDAFW Pump Abnormal Startup).

Justification

VA-20 powers the actuation logic for facility 2 AFAS and the actuation relays are energize-to-actuate. Loss of VA-20 means that facility 2 AFW components will have to be manually operated. The turbine driven AFW pump should not be used if a SGTR is in progress to prevent radiological contamination. The correct answer is to NOT start the TD AFW pump and close 2-FW-43A (AFW FRV to the #1 S/G) to prevent feeding the ruptured S/G. #2 S/G should be fed using both electric AFW pumps only.

Bank question 0065167 asked the applicants what the correct sequence would be if VA-10 was lost. This question was modified from losing VA-10 to losing VA-20. In addition, the previous question appeared to assume that a loss of VA-10 would fail open the FRV to the #1 S/G. This is not correct - loss of DV-10 causes 2-FW-43A to fail open. This modified question uses the previous bank question but corrects the earlier problems with that revision. Variations of the original distracters are used in the event that applicants memorized the answer to the bank question.

CHOICE [A] - NO
WRONG This was the previously correct answer to question 0065167 in the MP-2 bank - which was written as a loss of VA-10 instead of VA-20. It is not clear if this answer was ever truly correct. However, this answer is provided as a valid distracter for applicants who may have memorized the bank question. Using the turbine driven AFW pump to feed the #2 S/G when a SGTR is occurring is not recommended when both electric driven AFW pumps are fully functional. Selection of appendix 6 would be appropriate for starting the TDAFW pump and is consistent with the first part of the answer. .

CHOICE [B] - NO
WRONG Although this would result in feeding the #2 S/G, there would be no reason to manually initiate facility 2 AFW components if 2-FW-44 (AFW header cross-connect) was closed. In addition, using the TD AFW pump during a SGTR is not recommended. If the applicant thought that the loss of VA-20 would prevent a normal start of the TDAFW pump, then use of appendix 7 would be correct.

CHOICE [C] - YES
CORRECT The #1 AFW Reg valve (2-FW-43A) remains fully functional despite a loss of VA-20. This valve would fail open if DV10 was lost - which appears to be the previous correct answer to the bank question. Facility 2 AFW components would have to be manually operated because their actuation relay was deenergized when VA-20 lost power.

CHOICE [D] - NO
WRONG This distracter is incorrect because there is no reason to place the facility 2 hand switch in pull to lock and feeding the #2 S/G with the TDAFW pump would cause radiological problems - i.e. a release to the environment. Part 1 was an original distracter from the rev 1 version of this question. Use of appendix 6 would be appropriate if the TDAFW did not lose control power - which it does not with a loss of VA-20.

Question #: 1 | Question ID: 65167 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 4 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

References

1. AFW-00-C rev 5 chg 3, E.1.d. - Loss of Vital 120 VAC and E.3. - Operation of Terry Turbine AFP With SG Tube Leak.
2. EOP 2525 rev 24 page 16
3. AFW-00-C Figures 1 and 2

Comments and Question Modification History

Changed K/A from 061/A2.05 on original question and changed item 2 of choice 'C' from "Appendix 6 (TDAFW Pump Normal Startup)" to "Appendix 7 (TDAFW Pump Abnormal Startup)" to make choice 'C' clearly wrong (as written, the stated action is not "procedurally" wrong).

02/02/11; reworded four choices to improve readability, grammar and logic. - rlc.

NRC K/A System/E/A **System** 007 Reactor Trip

Number EA2.02 **RO** 4.3 **SRO** 4.6 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to determine or interpret the following as they apply to a reactor trip: Proper actions to be taken if the automatic safety functions have not taken place

Question #: **2** | Question ID: **1153616** | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. **3** | ☑ Selected for Exam | Origin: **Mod** | ☐ Past NRC Exam?

The plant was at 100% power when a Loss of Load caused the reactor to trip and the PORVs to briefly open.

Thirty minutes after the trip the following indications are noted:
- RCS pressure is 1850 psia and rising slowly.
- Quench Tank pressure is 35 psig and rising very slowly.
- CET temperatures are 534°F.

Which one of the following PORV discharge temperatures would indicate a PORV has not fully seated and why?

☑ **A** 281°F
The PORV discharge pressure is very close to the Quench Tank pressure; therefore, the discharge temperature will be close to the saturation temperature for this pressure.

☐ **B** 440°F
The PORV discharge pressure will drop to between RCS and Quench Tank pressure; therefore, the discharge temperature will be the saturation temperature for this pressure.

☐ **C** 534°F
The PORV discharge temperature is equal to the Reactor Coolant System temperature entering the pressurizer, which is close to the temperature indicated by the CETs.

☐ **D** 625°F
The PORV discharge temperature sensor is very close to the PORV; therefore, discharge temperature will be the saturation temperature for the RCS pressure.

Justification

A is correct. The PORV discharge pressure is very close to Quench Tank pressure; therefore, saturation temperature for the open PORV is the same as (or very close to) saturation temperature for the Quench Tank.

C is incorrect. The open PORV acts as a throttling device causing the downstream pressure to be considerably less than RCS pressure. Plausible: The examinee may believe that at low system pressure, the pressure downstream of a partially open PORV is the same as RCS pressure; therefore, discharge temperature is the same temperature as the RCS.

B is incorrect. The pressure downstream of the PORV is equal to the pressure in the Quench Tank.
Plausible: The examinee may believe that the pressure downstream of the PORV is equal to a pressure between Quench tank pressure and RCS pressure. If that were the case, then this is close to the saturation temperature for that enthalpy on the Mollier Diagram.

D is incorrect. PORV discharge temperature is not that close to the PORV to be held above saturation for the Quench tank pressure. Plausible: The examinee may believe that the PORV discharge temperature cannot be than the saturation temperature for CTMT conditions.

References

1. Steam Tables
2. Lesson Text, MCD-00-C, Mitigating Core Damage, Three Mile Island Accident

Comments and Question Modification History

02/02/11; Per validation, modified to be correct if Mollier diagram is used to answer. - rlc.

NRC K/A System/E/A **System** 008 Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)

Number AK3.02 **RO** 3.6 **SRO** 4.1 **CFR Link** (CFR 41.5,41.10 / 45.6 / 45.13)

Knowledge of the reasons for the following responses as they apply to the Pressurizer Vapor Space Accident: Why PORV or code safety exit temperature is below RCS or PZR temperature

Question #: N/A | Question ID: 53616 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 2 | ☐ Selected for Exam | Origin: Parent | ☐ Past NRC Exam?

PARENT

The plant was at 100% power when a Loss of Load caused the reactor to trip and the PORVs to open. Thirty minutes after the trip, RCS pressure is 1850 psia and the Quench Tank pressure is 45 psig.

Which one of the following PORV discharge temperatures would be indicated if a PORV is leaking by?

☐ **A** 250 degrees F

☐ **B** 274 degrees F

☑ **C** 293 degrees F

☐ **D** 625 degrees F

Justification

Sat. temp. for 45 psig (60 psia) is 292.7°F
#1 is sat. temp. for 30 psia (if mistakenly subtracted 15 from 45 to get psia)
#2 is sat. temp. for 45 psia
#4 is sat. temp. for 1850 psia

References

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 008 Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)

Number AA2.15 **RO** 3.9 **SRO** 4.2 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: ESF control board, valve controls, and indicators

Question #: 3 | Question ID: 1000004 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 1 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

In EOP 2525 the BOP is directed to check that at least one SG has BOTH:

a. 10 to 80% level.
b. MFW or TWO MDAFPs operating to restore level to 40 to 70%.

This action is credited in the Small Break LOCA analysis for which of the following reasons?

☐ **A** Ensures that stable, sub-cooled Natural Circulation can be established after coastdown of the Reactor Coolant Pumps.

☐ **B** Ensures Steam Generator tubes are re-covered for iodine scrubbing in the event of a subsequent Steam Generator Tube Rupture.

☐ **C** Ensures adequate inventory to maintain secondary side pressure such that Steam Generator tube sheet maximum D/P is NOT exceeded.

☑ **D** Ensures the Steam Generators are available to remove heat with the limited amount of inventory loss and injection flow.

Justification

D is correct, under worst case SBLOCA spectrum the injection flow is inadequate to prevent core uncovery. Reflux circulation removes heat w/o inventory loss;

A: SBLOCA analysis is for limiting cases, stable NC is not worst case;
Plausible: The examinee may believe that stable, subcooled NC must be maintained for a SBLOCA. While it is desired, it is NOT a requirement to be successful in mitigating the effects of a SBLOCA.

B: A factor for SGTR, but not a consideration for SBLOCA;
Plausible: The examinee may believe that SG tubes must be covered during a SGTR. While desirable for lodine scrubbing, it is NOT a requirement. In fact, it is desirable to maintain 40-45% SG level in the affected S/G for lodine scrubbing.

C: Max SG DP is only a concern for high RCS pressure.
Plausible: The examinee may believe that the maximum tube sheet D/P may be exceeded in a SBLOCA when, in fact this a bigger concern for an Excess Steam Demand.

References

EOP-2525 Tech. Guide; Pg. 15, St. #6 and also the step for "Perform a Controlled Cooldown"

Comments and Question Modification History

01/31/11; changed "D" from "**in support of a Small Break Loss of Coolant Accident**" to "**with the limited amount of inventory loss and injection flow**" - rlc

NRC K/A System/E/A **System** 009 Small Break LOCA

Number EK2.03 **RO** 3.0 **SRO** 3.3* **CFR Link** (CFR 41.7 / 45.7)

Knowledge of the interrelations between the small break LOCA and the following: S/Gs

Question #: 4 | Question ID: 1171905 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The plant has experienced a Large-Break LOCA inside containment.
All plant systems and components are functioning as designed and a Sump Recirculation Actuation Signal (SRAS) is expected to soon occur.

Which of the following describes the reason for procedurally directed actions, as they apply to the Large-Break LOCA and the flow path for sump recirculation?

☐ **A** RWST header isolation valves (CS-13.1A & CS-13.1B) must be closed to ensure the CTMT Spray pumps don't "short-cycle" their discharge back through the LPSI pumps.

☑ **B** SI minimum flow recirc valves, SI-659 and SI-660, must be positioned to "OPER" to prevent the flow of water back to the RWST and out the RWST atmospheric vent.

☐ **C** The CTMT Spray pumps must be secured to limit the amount of water drawn from the CTMT sump, thereby preventing loss of NPSH to the running HPSI pumps.

☐ **D** The LPSI pumps, after being secured by ESAS, must have their starting circuit overridden to prevent them from restarting on a post-SRAS LNP actuation.

Justification

B; CORRECT - This is in the initial actions when a SRAS is imminent and must be verified or manually accomplished to ensure a direct release to the environment does not exist.

A; WRONG - These are not the valves that would "short-cycle" the CS through the LPSI pumps. They are closed to provide an additional boundary to the existing check valves, which are designed for the stated concern, and to allow for subsequent re-filling of the RWST.
Plausible: the examinee may note that closing these valves is listed as a "Supplemental Actions" following a SRAS, but misinterprets the reason. There are valves controlled from the same panel that could cause short-cycling of CS, but they are normally closed.

C; WRONG - CS pumps are secured only if specific CTMT conditions exist, which are not mentioned in the stem.
Plausible: the examinee may note the stated reason is a valid one for securing the CS pumps, if indications of CTMT sump clogging exist.

D; WRONG - The LPSI pumps are automatically secured by ESAS and, based on the stem's amplifying information, do not have to be overridden.
Plausible: the examinee may confuse actions that are required to be taken during Shutdown Cooling operation to prevent an inadvertent bus voltage signal from affecting the LPSI pumps.

References

OP-2532 Tech. Guide, page 92, EOP Step Number 48 SRAS Initiation Criteria

Comments and Question Modification History

02/02/11; changed **"close"** in choice 'B' to **"OPER"** to better match actual switch position. - rlc.

NRC K/A System/E/A **System** 011 Large Break LOCA

Number EK3.08 **RO** 3.9 **SRO** 4.1 **CFR Link** (CFR 41.5 / 41.10 / 45.6 / 45.13)

Knowledge of the reasons for the following responses as the apply to the Large Break LOCA: Flowpath for sump recirculation

Question #: N/A | Question ID: 71905 | ☑ RO ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 2 | ☐ Selected for Exam | Origin: Parent | ☐ Past NRC Exam?

PARENT

The plant has experienced a Loss of Coolant Accident and the following conditions exist:
- Sump Recirculation Actuation has occurred.
- The Safety Injection Recirculation Header Isolation valves, 2-SI-659 and 660, remain open.

They are the only SRAS actuated components that have NOT automatically positioned, all other SRAS actuations have occurred as designed.

Which one of the following statements describes when and why these valves should be closed?

☑ **A** Immediately after other SRAS actuations have been verified, to prevent the unmonitored release of radiation through the recirc header, back to the RWST and out the RWST atmospheric vent.

☐ **B** Immediately after verifying 30 gpm minimum flow from each High Pressure Safety Injection (HPSI) pump, to ensure HPSI pumps do NOT overheat with the much hotter CTMT sump suction source.

☐ **C** Only after RWST header isolation valves (2-CS-13.1A & 2-CS-13.1B) are closed, to ensure the CTMT Spray pumps do NOT "short-cycle" their discharge back through the HPSI pumps.

☐ **D** Only after overriding and securing both LPSI pumps, to ensure the loss of minimum flow does NOT damage the HPSI and CTMT Spray pumps while these pumps continue to run.

Justification

A - Correct; EOP-2532, LOCA, dictates that 2-SI-659 & 2-SI-660 are verified closed before all of the actions mentioned in the distractors. These valves being open 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.

B - Wrong; Although closing the valves will isolate the HPSI minimum flow header, the accident analysis assumes these pumps will have sufficient flow to keep cool, even if being used post-SRAS.

C - Wrong; Although the RWST header isolation valves are closed, the Recirc Header isolation valves would NOT short-cycle CS through HPSI. There are valves on the discharge of the CS pumps that would do this, but they have NOT been analyzed for use and remain closed per administrative guidelines.

D - Wrong; LPSI pumps are automatically secured on a SRAS. However, if they were running due to operator action, they would challenge the other pumps based on suction flow (flow capacity of the suction strainers).

References

1. EOP-2532, St. 48.e, SRAS Initiation Criteria.
2. LP ESA-01-C, Pg. 20, d. SRAS Functions

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 013 Engineered Safety Features Actuation System (ESFAS)

Number A4.01 **RO** 4.5 **SRO** 4.8 **CFR Link** (CFR: 41.7 / 45.5 to 45.8)

Ability to manually operate and/or monitor in the control room: ESFAS-initiated equipment which fails to actuate

Question #: 5 | Question ID: 1100002 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant is operating at 100% power, with all systems and components available and functioning as designed.

Which one of the following malfunctions would require a plant trip and one or more RCP(s) to be immediately secured?

☐ **A** An ESAS malfunction causes both RCP Bleedoff containment isolation valves to close.

☑ **B** An RCP Vapor seal fails resulting in a valid Bleedoff Flow Lo alarm that remains locked in.

☐ **C** An RCP Upper seal fails resulting in a valid high bleedoff flow alarm that remains locked in.

☐ **D** The "C" RBCCW pump trips on overload with the "B" RBCCW pump aligned to bus 24C.

Justification

B - CORRECT; Failure of an RCP Vapor seal is the only seal failure that requires the plant be tripped and the RCP immediately secured.

A - WRONG; This does not require a trip because the Bleedoff relief valve would open and send flow to the Primary Drain Tank.
Plausible; Bleedoff would be isolated from any normal flow path, which would lend the examinee to believe it is blocked similar to the closure of an excess flow check valve.

C - WRONG; Pump trip is only required if the excess flow check valve closes on high bleedoff flow.
Plausible; High bleedoff flow is what causes the bleedoff flow check to close, which does require a pump trip.

D - WRONG; Under these conditions, the "B" RBCCW pump would be used to replace the "C" pump to prevent a plant trip on high RCP seal/bearing temperatures.
Plausible; Using the "B" pump under these conditions would violate Facility Separation and Tech. Specs. Prior to the installation of the "B" RBCCW Pump SIAS/LNP Block switch, a plant trip <u>was</u> required.

References

OP-2301C, R18C9, step 4.14.2

Comments and Question Modification History

01/31/11; Pat S. - "B" a bit confusing. Others OK. Also, delete the first "**at**" in the stem to correct grammar/typo. - rlc.

NRC K/A System/E/A **System** 015 Reactor Coolant Pump Malfunctions

Number AK2.07 **RO** 2.9 **SRO** 2.9 **CFR Link** (CFR 41.7 / 45.

Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: RCP seals

Question #: 6 | Question ID: 5000005 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 1 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

The plant is stable at 80% power with the following conditions:

- Letdown Flow Controller, HIC-110, is in MANUAL.
- Charging and letdown flow are balanced.

Then, an RCS leak occurs, causing Pressurizer level to lower at a rate of 2% every 10 minutes.
The US instructs the RO to stabilize Pressurizer level by adjusting the output of Letdown Flow Controller, HIC-110.

Final conditions; HIC-110 has been adjusted, Pressurizer level is now stable and there is NO makeup to the VCT.

Which one of the following describes the direction that the RO needed to adjust the output of HIC-110 to stabilize Pressurizer level, and at what rate will VCT level now lower?

☑ **A** Lowered the output, VCT now dropping at 4% every 10 minutes.

☐ **B** Lowered the output, VCT now dropping at 1% every 10 minutes.

☐ **C** Raised the output, VCT now dropping at 6% every 10 minutes.

☐ **D** Raised the output, VCT now dropping at 0.75% every 10 minutes.

Justification

Pressurizer volume per % indicated level => 66.44 gals/% (at 2250 psia)
VCT volume per % indicated level => 34 gals/%

CHOICE (A) - CORRECT
Controller output must be lowered to reduce letdown flow rate. Rate of VCT level decrease will be 1.954 (or approximately two) times the prior rate of pressurizer level decrease.

CHOICE (B) - INCORRECT
WRONG: the pressurizer volume per % indicated level is almost twice that of the VCT.
PLAUSIBLE: applicant may think the rate of VCT level decrease will be 1/2 that of the pressurizer.

CHOICE (C) - INCORRECT
WRONG: the controller output must be lowered to reduce letdown flow and the pressurizer volume per % indicated level is almost twice that of the VCT.
PLAUSIBLE: applicant may think controller output must be raised.

CHOICE (D) - INCORRECT
WRONG: the controller output must be lowered to reduce letdown flow and the pressurizer volume per % indicated level is almost twice that of the VCT.
PLAUSIBLE: applicant may think controller output must be raised.

References

1. CVC-00-C, "Chemical and Volume Control System" Lesson, Revision 9/2, C.5.c - Letdown Flow Control Valves.
2. OP-2304C, "Make Up (Boration and Dilution) Portion of CVCS", Revision 23/3 Section 4.6, "Batch Makeup to VCT" (Pg 25 of 98)
3. SP-2602A, "Reactor Coolant Leakage", Revision 6/1, Attachment 1, "RCS Pressure vs. Pressurizer Volume" (Pg 15 of 19)

Comments and Question Modification History

Question reworded to remove "fill-in" design and c

02/02/11; Per validation, deleted **"(0.2%/min.)"** from the stem as unnecessary info. -

NRC K/A System/E/A **System** 022 Loss of Reactor Coolant Makeup

Number AK1.03 **RO** 3.0 **SRO** 3.4 **CFR Link** CFR 41.8 / 41.10 / 45.3)

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

Question #: 7 | Question ID: 1183759 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The plant is in the process of cooling down for a refueling outage with the following conditions:

- SDC preparations are complete.
- 'A' & 'B' RCPs are still operating.
- Tc is being maintained at 265°F.

When RCS pressure is lowered to 250 psia, the 'SDC Suction Isolation Valves, 2-SI-651 and 2-SI-652, are opened, and Shutdown Cooling is placed in service.
The crew is attempting to stabilize RCS conditions in order to secure the remaining RCPs, when annunciators "2-SI-651 OPEN" and "2-SI-652 OPEN" alarm on C-01. (Windows C-9 and D-9 respectively)

What operator actions must be taken and why?

☐ **A** Close SDC Temperature Control Valve, 2-SI-657 and stop the cooldown to prevent a low temperature, over pressure condition in the SDC System.

☐ **B** Start a second LPSI Pump and lower RCS temperature to less than or equal to 300°F to prevent thermal stresses in the SDC system.

☑ **C** Either lower RCS pressure to less than 280 psia or close the SDC Suction Isolation Valves to prevent over pressurizing the SDC system.

☐ **D** Verify both "LT/OP Selector Switches" are set to LOW and that both PORVs are open to ensure SDC brittle fracture criteria are NOT exceeded.

Justification

C is correct. Annunciators "2-SI-651 OPEN and 2-SI-652 OPEN" alarm at an RCS pressure of 280 psia. The annunciators only provide a warning to the operator that the maximum SDC pressure will be exceeded if RCS pressure is allowed to rise to 300 psia.

A is incorrect. The alarms are not a function of low SDC temperature with high system pressure.
Plausible: The RCS is susceptible to low temperature/over pressure; therefore it would be logical to assume the SDC System is also susceptible to brittle fracture conditions. The examinee may believe that the alarm is a warning that the SDC System is approaching the low temperature over pressure limit.

B is incorrect. The alarms are not a function of SDC temperature.
Plausible: The design temperature limit on SDC is 300°F, however, the alarms are not associated with that limit. The examinee may believe that the alarm comes in to warn of a high SDC temperature, instead of high pressure.

D is incorrect. With the LT/OP Selector Switches set to LOW, the RCS is protected from a brittle fracture condition; however, the setpoint of 410 psia exceeds the design pressure for SDC (300 psia).
Plausible: The examinee may believe that the LT/OP setpoint protects the RCS and any system connected to it from a low temperature, over pressure condition..

References

1. ARP-2590A-035, R0C0; C-9 "SI-651 OPEN"
2. ARP-2590A-036, R0C0; D9 "SI-652 OPEN"

Comments and Question Modification History

01/31/11; Per Pat S. input, changed "**Just prior to securing**" to "**The crew is attempting to stabilize RCS conditions in order to secure**" in the stem. - rlc.

NRC K/A System/E/A **System** 025 Loss of Residual Heat Removal System (RHRS)

Number AK3.02 **RO** 3.3 **SRO** 3.7 **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 Residual Heat Removal System: Isolation of RHR low-pressure piping prior to pressure increase above specified level

Question #: N/A | Question ID: 83759 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 1 | ☐ Selected for Exam | Origin: Parent | ☐ Past NRC Exam?

PARENT

The plant has been shutdown for a refueling outage. SDC preparations are complete. 'A' & 'B' RCPs are still operating. The RO is depressurizing the RCS with Tc = 265° F. With PZR Press at 250 psia, the 'SDC Suction Isolation Vlvs', 2-SI-651 and 2-SI-652, are opened and the RCPs are secured.

Ten minutes after initiating SDC and securing RCPs, annunciators "2-SI-651 OPEN" and "2-SI-652 OPEN" alarm on C-01. (Windows C-9 and D-9 respectively)

What operator actions should be taken and why?

☐ **A** Ensure 2-SI-651 and 2-SI-652 automatically close to prevent over pressurizing the SDC system.

☐ **B** Immediately restart the RCPs to prevent an uncontrolled heat up of the RCS.

☐ **C** Ensure the "LT/OP Selector Switches" are set to LOW so the SDC system is NOT over pressurized.

☑ **D** Immediately initiate auxiliary spray to prevent over pressurizing the SDC system.

Justification

Annunciators "2-SI-651 OPEN and 2-SI-652 OPEN" alarm at 280 psia. The interlock that automatically closed 2-SI-652 upon receiving this annunciator was removed in the 1990s and annunciator only provides a prompt to the operator that the maximum SDC pressure

Question References not yet listed.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A System

Number	RO	SRO	CFR Link

Question #: 8 | Question ID: 1100004 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

During operation at 100% power, the following was noted:

- 'A' CEDM LO FLOW" alarm on C04
- CTMT Sump level rising
- RBCCW Surge tank level lowering
- "RBCCW SURGE TANK AUTO MAKEUP" alarm on C06/7

Which of the following actions are required?

☑ **A** Trip the plant, secure "A" & "C" RCPs, and then close the "A" RBCCW Header supply and return isolations to CTMT.

☐ **B** Trip the plant, secure "A" & "C" RCPs and the "A" & "C" CAR Fans, then close "A" RBCCW Header supply and return isolations to CTMT.

☐ **C** Close the "A" RBCCW Header CTMT isolations to the "A" CEDM cooler and start the standby CEDM cooling fan.

☐ **D** Close the "A" RBCCW Header CTMT isolations to the "A" CEDM cooler and verify "B" & "D" CAR Fans are operating in "fast".

Justification

A - CORRECT; This is indicative of a major leak on the "A" CEDM Cooler. All three CEDM Coolers are supplied by the "A" RBCCW header and are on the same line in CTMT that supplies the "A" & "C" RCPs. The valves specific to the CEDM Coolers are located in CTMT and can not be closed in time to prevent the RCPs from being affected by the loss of RBCCW on the "A" header (surge tank level is lowering with full makeup). Therefore, in order to secure expeditiously isolate the leak, the CTMT isolation valves for that RBCCW line must be closed. This will require the "A" & "C" RCPs be secured and, therefore, the plant must first be tripped.

B - WRONG; The "A" RBCCW header isolation valves that isolate RBCCW to the CEDMs also isolate RB flow to the "A" & "C" RCPs, but not the "A" & "C" CAR Fans.
Plausible: Examinee may believe all equipment vital for plant operation supplied by an RBCCW header is isolated by the applicable header isolation valves (as implied by the simplified one-line system training drawing).

C - WRONG; Isolating RBCCW to one CEDM cooler will isolate flow to all of them. Therefore, there is no purpose in starting the standby fan.
Plausible; Examinee may believe the CEDM Coolers are supplied by a different supply, like the vast majority of the heat loads on the RBCCW system. Although the coolers have the capability to be supplied from different headers, they are commonly all supplied from the "A" header only.

D - WRONG; Although the "A" & "C" CAR Fans will certainly be impacted if the "A" RBCCW header is lost, this action will not mitigate the major problem with losing the header, specifically, damage to the "A" & "C" RCPs.
Plausible; Examinee may believe closing RBCCW isolation valves for CEDM Coolers will either directly affect RBCCW to the applicable CAR Fans or, because the CAR Fans are Safety System components used to prevent exceeding CTMT design limits in the event of a LOCA or ESD, performing actions to safeguard their operation is more critical than potential damage to the RCPs.

References

AOP-2564, R4C3; Section 10, "Response to RBCCW Piping Rupture"

Comments and Question Modification History

02/02/11; Per validation, fixed bullets in stem. - rlc.

NRC K/A System/E/A **System** 026 Loss of Component Cooling Water (CCW)

Number AA2.03 **RO** 2.6 **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 valve lineups necessary to restart the CCWS while bypassing the portion of the system causing the abnormal condition

Question #: 9 | Question ID: 1180008 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The following initial plant conditions exist:

- 100% steady-state
- Channel "Y" Pressurizer Level and Pressure Control set up as the controlling channels.
- Forcing sprays with 4 sets of backup heaters energized.
- Channel "Y" Pressure Controller setpoint at 2200 psia, maintaining pressure at 2250 psia.

Then, the Channel "Y" High Pressurizer Pressure bistable (setpoint of 2350 psia), fails to the "actuated" mode. All other pressurizer control system components are functioning normally and respond as designed to the relay actuation.

Which of the following describes the change in indications that would be seen, if NO operator actions where taken?

☑ **A** The pressurizer backup heaters would deenergize and RCS pressure would lower causing proportional heater output to rise and stabilize pressure at approximately 2200 psia.

☐ **B** The pressurizer backup heaters would deenergize and the lower pressure controller setpoint would cause RCS pressure to lower until the plant trips on low RCS pressure.

☐ **C** All Pressurizer heaters would deenergize and RCS pressure would lower to 2200 psia, causing the backup heaters to reenergize and maintain RCS pressure between 2200 psia and 2225 psia.

☐ **D** All Pressurizer heaters would deenergize and spray valve bypass flow and general heat loss would cause RCS pressure to continue to lower until the plant trips on low RCS pressure.

Justification

A - CORRECT; When the High Pressure bistable/relay triggers, it trips the backup heaters and prevents all other control signals from re-energizing them. The bistable/relay is powered by a non-vital bus and fails to the "actuate" mode when de-energized. Because of this, it trips only the backup heaters when it triggers and has NO effect on the proportional heaters. Therefore, the proportional heaters will ramp up in output as pressure lowers to the controller setpoint of 2200 psia and stabilize pressure at the setpoint value.

B - Wrong; Because the bistable/relay has NO effect on the proportional heaters, they will respond to the lowering RCS pressure as designed.
Plausible; The low RCS pressure pre-trips are approximately 50 psi below the present controller setpoint, the proportional heaters are designed to go to maximum output at 25 psi below setpoint and the spray valves are designed to fully close 50 psi above setpoint. The examinee may confuse the various setpoint values and believe the system will not turn pressure before the trip value with the abnormally low controller setpoint.

C - Wrong; With the bistable/relay triggered, the backup heaters are unavailable, regardless of operator or system actions.
Plausible; Because the setpoint for the relay that automatically turns on the backup heaters on low RCS pressure is below the DNB Tech. Spec. limit of 2225 psia, the examinee may believe that this "turn on" relay would, therefore, override the "trip" relay.

D - WRONG; The proportional heaters are still available and would be able to stabilize pressure at the controller setpoint.
Plausible; The examinee may believe all heaters must be tripped by this relay as the setpoint is only approximately 45 psi below the high pressure trip value, which also opens both PORVs.

References

1. OP-2204, R22C1; Attachment 3, Pressurizer Pressure Control Program
2. PLC-01-C, R4; Section C.17.b - Pressurizer Pressure Bistables, Design and Operating Characteristics

NO Comments or Question Modification History at this time.

NRC K/A System/E/A System 027 Pressurizer Pressure Control System (PZR PCS) Malfunction

Number AA1.04 **RO** 3.9* **SRO** 3.6* **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: Pressure recovery, using emergency-only heaters

Question #: N/A | Question ID: 8000008 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☐ Past NRC Exam?

PARENT

The following initial plant conditions exist:

- 100% steady-state
- Channel "X" Pressurizer Level and Pressure Control set up as the controlling channels
- Forcing sprays with 4 sets of backup heaters
- Channel "X" Pressure Controller setpoint at 2200 psia, maintaining pressure at 2250 psia

Then, VR-21 deenergizes due to a problem with its static switch.

Which of the following describes the status of the applicable components, assuming NO operator actions have been taken?

☐ **A** Only Facility Two pressurizer heaters have deenergized, causing RCS pressure to lower and diminish spray flow.

☐ **B** Only the pressurizer backup heaters would be deenergized and pressure would stabilize at approximately 2200 psia.

☐ **C** All Pressurizer heaters are deenergized, RCS pressure would lower to 2200 psia causing the backup heaters to reenergize.

☑ **D** All Pressurizer heaters are deenergized and spray valve bypass flow would cause RCS pressure to continue to lower.

Justification

D - Correct; The Pressurizer Heater Selector switch is normally in the "Both" position, which means a loss of VR-11 OR VR-21 will cause all PZR heaters to deenergize due to the failure of the heater low level cutout circuit. The recovery of the heaters requires the operators to de-select the failed/de-energized circuit (select Ch. "X" only) and reclose both Proportional heater breakers.
A - Wrong; This would be true if it was a non-vital 480 VAC bus that was lost (I.e.; 22A - 22D).
Plausible: The examinee may believe that the loss of VR-21 results in the loss of only the Pressurizer heaters powered from Facility 2.
B - Wrong; The loss of VR-21 trips all heaters because the heater low level cutout is designed to protect even the vital, proportional heater groups.
Plausible: The examinee may believe that the loss of VR-21 results in the loss of only the non-vital Pressurizer heaters powered from Facility 2.
C - Wrong; With VR-21 deenergized, the backup heaters are unavailable, regardless of operator or system actions. This is because the loss of VR-21 causes the High Pressurizer Pressure heater trip to fail in the "triggered" mode, which prevents the Backup heaters from being re-energized by operator OR control system action.
Plausible: The examinee may believe that the Backup heaters will reset and energize when Pressurizer pressure reaches the backup signal setpoint.

References

AOP-2504B, R-3, C-11, St. 3.2
Loss-Of-Control-Power Operator Aid, R-1, C-0

Comments and Question Modification History

Reworded Distractor 'C' slightly to state that 'RCS pressure lowering to 2200 psia would cause the backup heaters to reenergize. Provides clarity to distractor.
Reworded Answer 'D' to state that pressurizer spray bypass flow would cause RCS pressure to lower. Provides clarity and makes the correct answer 'more correct.' 11/11/08

NRC K/A System/E/A **System** 027 Pressurizer Pressure Control System (PZR PCS) Malfunction

Number AA1.01 **RO** 4.0 **SRO** 3.9 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: PZR heaters, sprays, and PORVs

Question #: 10 | Question ID: 1140006 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The plant was operating at 100% power, MOL, with all conditions normal when a malfunction caused a Turbine trip. The reactor failed to trip automatically or by use of the manual trip push buttons; however, the Diverse Scram System (DSS) functioned as designed shortly after the Turbine trip to mitigate the ATWS.

Which of the following describes the response of reactor power to both the Turbine trip and the operation of the DSS?

☐ **A** Initially rise due to the lower production of Xenon and higher RCS pressure, then drop quickly due to the injection of boric acid.

☐ **B** Initially rise due to the lower production of Xenon and higher RCS pressure, then drop quickly due to the insertion of the CEAs.

☐ **C** Initially lower due to the effects of the moderator and fuel temperature coefficients, then drop quickly due to the injection of boric acid.

☑ **D** Initially lower due to the effects of the moderator and fuel temperature coefficients, then drop quickly due to the insertion of the CEAs.

Justification

D - CORRECT; When the turbine trips and the Reactor does NOT, RCS temperature will rise due the sudden decrease in heat removal. This will also cause a rise in the Fuel temperature. The rise in both fuel and moderator temperature will each add negative reactivity causing Reactor power to lower. The DSS is designed to de-energize the CEDMs by an alternative method (from RPS) and cause the insertion of all CEAs.

A - WRONG; Reactor power will NOT rise. RCS Pressure will rise and Xenon production will lower slightly inserting a small amount of positive reactivity, but it will be insignificant compared to the negative reactivity inserted due to the RCS temperature rise. Also, the DSS will NOT initiate boron injection.
Plausible: The examinee may believe that the positive reactivity inserted by the significant rise in RCS pressure and the lower Xenon production will overshadow the negative Reactivity inserted by the rise in RCS temperature. Additionally, the examinee may believe that the DSS results in the rapid injection of Boric Acid.

B - WRONG; Although the DSS does insert the CEAs through an alternate means, reactor power will NOT rise initially. As power is reduced due to the rise in temperature, Xenon production will lower, but will be negligible. RCS Pressure will rise and insert a small amount of positive reactivity, but it will be insignificant.
Plausible: The examinee may believe that the positive reactivity inserted by the significant rise in RCS pressure and the lower Xenon production will overshadow the negative Reactivity inserted by the rise in RCS temperature resulting in a rise in Reactor power, which will stop rising when CEAs are inserted.

C - WRONG; Although power will lower due to the effects of MTC and FTC, the DSS does NOT insert negative reactivity by boron injection.
Plausible: The examinee may believe that the DSS results in the rapid injection of Boric Acid.

References

ARP-2590C-101, R0C0; D-13, "Diverse RX Trip Actuated"

Comments and Question Modification History

01/04/11; Modified stem question per Sandy Doboe input.

NRC K/A System/E/A **System** 029 Anticipated Transient Without Scram (ATWS)

Number EK1.02 **RO** 2.6 **SRO** 2.8 **CFR Link** (CFR 41.8 / 41.10 / 45.3)

Knowledge of the operational implications of the following concepts as they apply to the ATWS: Definition of reactivity

Question #: N/A | Question ID: 4000006 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☐ Past NRC Exam?

P A R E N T

The plant was operating at 100% power with all conditions normal when a malfunction caused a Turbine trip. The Reactor failed to trip automatically or by use of the manual trip push buttons; however, the CEDM MG breakers were opened by the Diverse Scram circuitry shortly after the Turbine trip. All other components operated as designed. Which of the following statements describes the status of Reactor power just prior to the CEDM MG breakers opening and the reason for the change in Reactor power?

☐ **A** Rise due to effects of the void coefficient and the moderator temperature coefficient

☐ **B** Rise due to the effects of Xenon burnout and the fuel temperature coefficient

☐ **C** Lower due to the effects of Xenon building in and the void coefficient

☑ **D** Lower due to the effects of the moderator and fuel temperature coefficients

Justification

D is correct. When the turbine trips and the Reactor does NOT, RCS temperature will rise due the sudden decrease in heat removal. This will also cause a rise in the Fuel temperature. The rise in both fuel and moderator temperature will each add negative reactivity causing Reactor power to lower. A is incorrect. Reactor power will NOT rise. The void coefficient will add a slight (insignificant) amount of negative reactivity and will only cause a slight lowering of Reactor power. The moderator temperature coefficient will add negative reactivity causing Reactor power to lower. B is incorrect. Reactor power will NOT rise. Xenon will NOT burn out, especially in this short a period of time. The fuel temperature coefficient will add negative reactivity causing Reactor power to lower. C is incorrect. Although power will lower, Xenon will NOT build in and will have NO impact on the magnitude of the power change. Additionally, the void coefficient may add a slight, but insignificant, amount of negative reactivity to lower power.

Question References not yet listed.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 029 Anticipated Transient Without Scram (ATWS)

Number EK1.02 **RO** 2.6 **SRO** 2.8 **CFR Link** (CFR 41.8 / 41.10 / 45.3)

Knowledge of the operational implications of the following concepts as they apply to the ATWS: Definition of reactivity

Question #: 11 | Question ID: 1100006 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant automatically tripped due to a Steam Generator Tube Rupture on #1 Steam Generator with a subsequent loss of Offsite Power. The crew successfully completed EOP 2525, Standard Post Trip Actions. The affected Steam Generator has been isolated per EOP 2534, Steam Generator Tube Rupture. The subsequent cooldown (after lowering both hot leg temperatures to less than 515°F) has been continuing for the past 30 minutes at approximately 70°F/hr.

The difference between Loop 1 Th and Loop 2 Th is 12°F
RCS pressure is 600 psia
CETs are reading 450°F

Based on the above information, which of the following actions is procedurally required and what is the basis for this action?

☐ **A** Raise the cooldown rate to between 80°F/hr and 100°F/hr.
To ensure that Shutdown Cooling is placed in service within the required time after the event.

☐ **B** Lower RCS pressure to between 470 psia and 500 psia.
To minimize the volume of water leaking from the Reactor Coolant System to the affected Steam Generator.

☑ **C** Lower the cooldown rate to between 10°F/hr and 25°F/hr.
To keep the loops coupled and ensure the isolated Steam generator is adequately cooled and depressurized.

☐ **D** Raise RCS pressure to between 850 psia and 900 psia.
To eliminate voiding and ensure that natural circulation flow is adequate to continue the cooldown.

Justification

C is correct. A difference of more than 10°F in loop hot leg temperatures is indication of the S/Gs becoming 'uncoupled'. As a result, the isolated S/G becomes a heat source for the RCS and the cooldown begins to stall (i.e., core heat removal is NOT adequate). The proceduralized method for ensuring the isolated S/G is being adequately cooled is to slow the cooldown and allow the isolated S/G to equalize with the intact S/G.

A is incorrect. Raising the cooldown rate will cool and depressurize the intact S/G; however the isolated S/G will NOT cool down and will prevent depressurizing the RCS.
Plausible: The initial direction is to perform the cooldown at the maximum controllable rate. The Tech spec limit for an RCS cooldown is 100°F/hr. If the examinee doesn't realize there is a different procedural limit on the cooldown rate for maintaining the loops coupled, then he/she may believe that the Tech Spec cooldown rate is the only limit. Additionally, there is a time limitation of 16 hours for placing SDC in service after a SGTR.

B is incorrect. Lowering RCS pressure will allow more safety injection flow, but will also lower subcooling below the low limit of 30°F.
Plausible: EOP 2534 directs the crew to maintain RCS pressure as low as possible to reduce or eliminate the primary to secondary leakage. It also directs the crew to maintain RCS pressure within the P/T limits (30°F subcooled). If the examinee believes that eliminating the leakage is a higher priority than maintaining parameters within the P/T limit, then he/she may believe that RCS pressure should be maintained as close to saturation as possible.

D is incorrect. A head void is likely with no RCPs operating. Raising RCS pressure will help eliminate the void, but will NOT improve the cooldown on the affected S/G and will only cause the leakage from the RCS to the affected S/G to rise.
Plausible: The examinee may believe that the loop differential temperature is caused by head voiding which is affecting natural circulation flow.

References

1. EOP-2534, R25, Pg 27, Note 2
2. EOP-2534, R25, Pg 49, St 58.a.2)

Comments and Question Modification History

12/03/10; Chip Griffin: Add length of time that cooldown has been ongoing.

NRC K/A System/E/A System 038 Steam Generator Tube Rupture (SGTR)

Generic K/A Selected

NRC K/A Generic System 2.4 Emergency Procedures /Plan

Question #: 11 | Question ID: 1100006 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

Number 2.4.18 **RO** 3.3 **SRO** 4.0 **CFR Link** (CFR: 41.10 / 43.1 / 45.13)

Knowledge of the specific bases for EOPs.

Question #: 12 | Question ID: 4071648 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The plant was operating normally at 100% power when the "B" Main Steam Header ruptured in Containment.

The Auxiliary Feedwater Actuation Signal (AFAS) has NOT yet actuated.

Under the existing conditions, which of the following actions must be taken to help mitigate this event and why?

☐ **A** Place the "A" and "B" Motor-Driven AFW pump switches in "Pull-To-Lock" to prevent water hammer in the "B" S/G due to the addition of cold feedwater with a low S/G level.

☑ **B** Place both S/G Auto AFW Override switches in "Pull-To-Lock" to prevent challenging Containment parameters due the addition of feedwater to the affected SG.

☐ **C** Shift both S/G AFW regulating valve controllers to "Manual" and "Closed" to ensure that feedwater flow will be added slowly to limit the cooldown of the RCS.

☐ **D** Momentarily place both AFW regulating valve "RESET NORM OVRD" switches to "OVRD" to ensure that AFW can be manually controlled once it automatically initiates.

Justification

B - CORRECT; Placing both S/G Auto AFW Override switches in "Pull-To-Lock" will prevent auto aux feed from initiating and feeding the affected SG, which would add an excessive amount of energy to the CTMT environment as the added water boils off.

A - WRONG; The "A" and "B" motor-driven aux feed pump hand switches do not have the same Pull-To-Lock feature as the AFW System facility hand switches. Also, water hammer, although a potential concern, is not the overriding problem with continuing to feed the ruptured SG.
Plausible; The examinee may recall that feeding a hot SG with cold feed water when level is low has been known to destroy SG feed rings due to water hammer.

C - WRONG; Shifting the AFW Regulating valves to "Manual" and "Closed" at this time will NOT prevent the valve from automatically opening.
Plausible; the examinee may believe manual control will prevent the valve from fully opening automatically and allow for a slower feed rate, which would accomplish the desirable goal of limiting the RCS cooldown.

D - WRONG; Attempting to overriding the AFW regulating valve at this time will NOT prevent the valve from automatically opening.
Plausible; the examinee may believe that use of this switch will allow for manual control of AFW because that is the intended purpose of the switch.

References

1. OP-2260, R9C2, EOP-2525 Critical Tasks/Operator Credited Actions #2.
2. EOP-2525, R23, Contingency Actions 6.b.2 2)
3. EOP-2536, R24, EOP-2525 Critical Tasks/Operator Credited Actions #1.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** E05 Excess Steam Demand

Number EK3.2 **RO** 3.3 **SRO** 3.8 **CFR Link** (CFR: 41.5 / 41.10, 45.6, 45.13)

Knowledge of the reasons for the following responses as they apply to the (Excess Steam Demand): Normal, abnormal and emergency operating procedures associated with (Excess Steam Demand).

Question #: N/A | Question ID: 71648 | ☑ RO ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 2 | ☐ Selected for Exam | Origin: Parent | ☑ Past NRC Exam?

PARENT

The plant was operating normally at 100% power, when the following events occurred:

- Pressurizer Pressure, Level, and Reactor Coolant (RCS) Cold Leg Temperature (Tc) start dropping rapidly
- Reactor trips
- Main Steam Isolation (MSI) and Safety Injection Actuation Signal (SIAS) occur
- Reactor Coolant Pumps (RCPs) are secured
- Loop 2 Tc and Steam Generator (S/G) pressure are decreasing much faster than Loop 1 Tc and S/G pressure.
- Auxiliary Feedwater Actuation Signal (AFAS) has NOT actuated
- Containment pressure and temperature are increasing

Which of the following actions must be taken on Panel C-05 in accordance with EOP-2536, "Excess Steam Demand Event" to mitigate this event?

☐ **A** Place #2 S/G Auxiliary Feedwater Isolation Air Assisted Check Valve Switch to CLOSE.

☐ **B** Shift #1 and #2 Auxiliary Feedwater Regulating Valve Controllers to MANUAL and CLOSED.

☑ **C** Place #1 and #2 SG Auto Permissive OVERRIDE/MAN/START RESET Switches to PULL-TO-LOCK.

☐ **D** Shift #2 S/G Auxiliary Feedwater Regulating Valve RESET/NORM/OVRD Switch momentarily to OVRD.

Justification

CHOICE (A) - NO
WRONG: The air assisted check valves are designed to provide containment isolation in the event of an accident inside containment. These valves are are 6 inch swing checks that will prevent a reversal of flow. Normal AFW flow will open the valves.
VALID DISTRACTOR: an EOP Step (EOP-2536, Step 9.L, Pg 12 of 62) directs closing this valve in the event of a steam line break. Applicant may think that closing this valve will prevent AFW from reaching the SG.

CHOICE (B) - NO
WRONG: An auto actuation signal will open the AFW feed regulating valves even in the manual loading stations are in MANUAL and CLOSED.
VALID DISTRACTOR: applicant may assume that the valve will not automatically open when in MANUAL.

CHOICE (C) - YES
The AFW feed regulating valves will be closed until AFAS is actuated. Placing these switches in PULL-TO-LOCK prior to AFAS blocks the automatic initiation signal that opens the AFW feed regulating valves. (AFW-00-C, Pg 19 of 56)

CHOICE (D) - NO
WRONG: The RESET NORM OVRD switch will not prevent feeding the SG if Auto AFW trips after the RESET NORM OVRD was momentarily (spring return to normal) in OVRD.
VALID DISTRACTOR: applicant may think that once overridden, the valve will not react to an auto actuation signal until this same switch is taken to RESET.

References

1. AFW-00-C, "Auxiliary Feedwater System" Lesson, Revision 5 (Pg 19, 20 of 56)
2. EOP-2536, "Excess Steam Demand Event", Revision 20 (2/27/01) (Pg 12 of 62)

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** E05 Excessive Heat Transfer

Number EA1.1 **RO** 3.9 **SRO** 4.2 **CFR Link** (CFR: 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the (Excessive Heat Transfer) Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question #: 13 | Question ID: 1171926 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The plant tripped from 100% power due to loss of all feedwater approximately 45 minutes ago.
The following conditions now exist:

- Buses 25A and 25B are deenergized due to a loss of the grid (state wide blackout).
- Bus 24E is aligned to Bus 24C.
- Bus 24C is deenergized and the "A" Emergency Diesel Generator is failed and unavailable.
- "B" AFW pump and the Turbine Driven AFW pump have both failed and are unavailable.
- #1 S/G level is at 99 inches and will drop to 70 inches in 8 minutes.
- #2 S/G level is at 130 inches and will drop to 70 inches in 12 minutes.
- The RO is continuing to evaluate various annunciators on C-01.
- The BOP is attempting to reenergize bus 24C from Unit 3, estimates 10 minutes to reenergize 24C.
- The US is performing applicable duties and NO other licensed personnel are available in the Control Room.
- All other plant systems and components are operating or available as designed.

Which of the following is required per the applicable procedures?

☑ **A** The RO needs to immediately stop what he/she is doing and initiate Once-Through-Cooling before SG level reaches 70".

☐ **B** The RO needs to immediately stop what he/she is doing and assist with the restoration of power on Facility 1 before SG level reaches 70".

☐ **C** The RO needs to complete his/her present task and then initiate Once-Through-Cooling as time permits.

☐ **D** The BOP needs to complete his/her present task and then initiate Once-Through-Cooling as time permits.

Justification

A - CORRECT; Note prior to step 5 of EOP 2537 states:
Once through cooling should be initiated prior to SG wide range level reaching 70 inches if any of the following exists:
1. Main or Auxiliary Feedwater is NOT expected to be restored.
2. Less than two trains of HPSI, PORVs, or ADVs are available.

Additionally, OP 2260 EOP User's Guide states that OTC should be initiated at 100" to ensure it is complete by the time S/G level reaches 70".

B - WRONG; The loss of power does not have a critical effect on the Vital Auxiliary Safety Function because facility 2 is powered.
Plausible; The restoration of power is part of the Vital Auxiliaries safety function, which is a higher safety function than RCS/Core Heat Removal. Based on this, the examinee may feel that power restoration is greatest concern under these conditions.

C - WRONG; Although Once-through-Cooling must be initiated early to ensure adequate heat removal with only one HPSI available, the US must maintain an oversight position to monitor the restoration and maintenance of plant Safety Functions.
Plausible; The examinee may believe that both tasks being performed by the board operators are critical as they will restore Safety Functions and, therefore, should not be delayed in their completion.

D - WRONG; Once Through Cooling must be initiated early to ensure adequate heat removal with only one HPSI Pump injecting.
Plausible; As Once-Through-Cooling involves the deliberate rupturing of the RCS barrier, the examinee may believe that with 24C expected to be restored (and thereby a source of feedwater) before both S/Gs drop below 70", it is preferable to allow this task to be completed.

References

1. EOP-2537, R21; Note prior to Step 5.
2. OP-2260, R9C2; EOP-2537 General Expectations #1

Comments and Question Modification History

01/31/11; Per Larry S. input, changed **"should"** in each choice to **"needs to"**. - rlc.

02/02/11; Per validation, changed what the RO was initially doing in the stem to be more realistic. Reworded choice 'A' to make it correct. Slightly modified choice 'B' to better align with choice 'A'. - rlc.

NRC K/A System/E/A **System** E06 Loss of Feedwater

Number EK3.4 **RO** 3.2 **SRO** 3.7 **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 Feedwater): RO or SRO function within the control

room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

All Exam Questions Designated RO or SRO (Includes "Parents" + "Originals")

Question #: N/A | Question ID: 8071926 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☑ Past NRC Exam?

PARENT

Following a trip from 100% power due to loss of all feedwater, the following plant conditions exist:

- Buses 25A and 25B are deenergized due to a failure to automatically fast transfer.
- Bus 24E is aligned to Bus 24C.
- Bus 24C is deenergized; the associated D/G will NOT start (PEO dispatched).
- "B" Aux Feedwater pump breaker tripped on fault. (PEO dispatched)
- The Terry Turbine tripped on overspeed and will NOT reset. (PEO dispatched)
- The Condensate System is NOT in operation.
- #2 S/G level is 130 inches and lowering.
- #1 S/G level is at 100 inches and lowering.
- Trending indicates #1 SG level will be at 70 inches within the next 10 minutes.
- All other conditions are as expected.

Early implementation of Once-Through -Cooling ________________.

☐ **A** will NOT be necessary at this time because Feedwater may be restored prior to reaching 70 inches in either S/G.

☐ **B** should be initiated now because the Condenser Steam Dumps are NOT available for heat removal.

☐ **C** will NOT be necessary at this time because both Atmospheric Dump Valves are available for heat removal from the S/Gs.

☑ **D** should be initiated now because only one train of HPSI is available for heat removal with the PORVs.

Justification

D is correct; Note prior to step 5 of EOP 2537 states:
Once through cooling should be initiated prior to SG wide range level reaching 70 inches if any of the following exists:
1. Main or Auxiliary Feedwater is NOT expected to be restored.
2. Less than two trains of HPSI, PORVs, or ADVs are available.
3. NO Charging Pumps are available.

Additionally, OP 2260 EOP User's Guide states that OTC should be initiated at 100" to ensure it is complete by the time S/G level reaches 70".
A is incorrect; Although it is a possibility that feedwater may be restored prior to reaching 70 inches in either S/G, with only one HPSI available, Once-through-Cooling must be initiated early to ensure adequate heat removal.
B is incorrect; Although the Condenser Steam Dumps are NOT available due to the loss of Condensate (MSIVs are closed), this is NOT a criteria for early initiation of Once-Through-Cooling.
C is incorrect; Although both ADVs are available for heat removal at this time, the loss of feed to the S/G will result in a loss of heat removal from the S/Gs when inventory is depleted. Once Through Cooling must be initiated early to ensure adequate heat removal with only one HPSI Pump injecting.

References

EOP 2537, Loss of All Feedwater, note prior to Step 5.

Comments and Question Modification History

Changed #2 SG level from 235 inches to 150 inches and #1 SG level from 150 inches to 110 inches. This is to ensure the examinee realizes that 'early initiation' should begin now and not wait to see if components can be restored before 'early initiation' is attempted. 11/11/08

Lowered SG levels to 130" and 100", per NRC comments, to more effectively fit the knowledge requirements of an RO. 01/13/2009
The following objectives signify that an RO is required to know the conditions, actions, and bases for the initiation of once through cooling:
MB-5960, LOIT Given a set of plant conditions concerning a loss of all feedwater, determine if criteria for the following are met as specified in EOP 2537, Loss of All Feedwater:
B. initiation of once through cooling
MB-05961, LOIT Describe the condition dependent actions and their bases for the following as specified in EOP 2537, Loss of All Feedwater:
A. initiation of once through cooling
B. fullyimplementingg once through cooling via PORVs.

NRC K/A System/E/A **System** E06 Loss of Feedwater

Number EA2.2 **RO** 3.0 **SRO** 4.2 **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 Loss of Feedwater.

Question #: N/A | Question ID: 8071926 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☑ Past NRC Exam?

Question #: 14 | Question ID: 1100007 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 1 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant has tripped from 100% power on a sudden loss of condenser vacuum when the Main Condenser Boot Seal ruptured. On the plant trip, a total loss of off-site power occurred due to fault on the RSST.

Assume all systems and components are functioning as designed.
Assume no operator actions have been taken or will be taken.

For the next 15 minutes, starting at the time of trip, how will **Tavg** respond?

☐ **A** Initially spikes up, then quickly lowers and stabilizes on automatic operation of the Main Turbine Bypass Valve.

☑ **B** Initially spikes up, then quickly lowers and stabilizes on automatic operation of the Atmospheric Dump Valves.

☐ **C** Drop suddenly, then quickly rises and stabilizes on automatic operation of the Atmospheric Dump Valves.

☐ **D** Drop suddenly, then quickly rises and stabilizes on automatic operation of the Main Steam Safety Valves.

Justification

B - CORRECT; Tavg will initially rise due to the turbine tripping before the reactor. When condenser vacuum drops below 15", the Condenser Dump Valves are interlocked closed. This only leaves the ADVs to modulate as required to maintain Tavg with decay heat loads.

A - WRONG; The Turbine Bypass valve will fail closed when condenser vacuum degrades below 15". With the plant tripping due to a ruptured boot seal, condenser vacuum should drop below 15" very quickly.
Plausible; The examinee may believe that recent control power changes would allow operation of the Bypass Valve with a loss of off-site power. This change to control power prevents the "loss of vacuum" inhibit from triggering in error due to a loss of off-site power.

C - WRONG; The reactor trip was caused by the turbine trip, which will cause an initial rise in Tavg.
Plausible; The examinee may focus on the loss of power and recognize (correctly) that the power loss will not immediately prevent the condenser dump valves from opening. With all six dump valves opening on a turbine trip caused by a reactor trip, Tavg would normally go down quickly.

D - WRONG; The ADVs will still be available 15 minutes after the loss of power to stabilize Tavg.
Plausible; The examinee may not recall that recent changes made to the steam dump control power will allow a normal system response of the ADVs.

References

RRS-01-C, R4C4, Pgs. 16-20, "2. Abnormal Operation"

Comments and Question Modification History

01/05/11; Revised question stem and choices based on Sandy Doboe's review.

01/31/11; Pat S. - add **"quickly"** to each choice. - rlc

NRC K/A System/E/A **System** 056 Loss of Offsite Power

Number AA2.32 **RO** 4.3 **SRO** 4.3 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Transient trend of coolant temperature toward no-load T-ave

Question #: 15 | Question ID: 1100009 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant is operating at 100% power when VA-10 is lost due to a failure of the panel's main breaker.

If #1 Steam Generator (S/G) level starts to slowly lower, how does AOP-2504C, Loss of VA-10 direct S/G level be controlled.

☐ **A** Manual control of S/G Feed Pump speed and manual control (C05) of #2 Main Feed Regulating Valve.

☐ **B** Automatic control of S/G Feed Pump speed and Local-Manual control of #1 Main Feed Regulating Valve.

☐ **C** Manual control of #1 S/G Feed Pump speed and manual control (C05) of #1 Main Feed Regulating Valve.

☑ **D** Manual control of S/G Feed Pump speed and Local-Manual control of #1 Main Feed Regulating Valve.

Justification

D - CORRECT; The #1 FRV will lock up "as-is" on a loss of VA-10. The procedure directs adjusting SGFP speed in manual to control feedwater flow and S/G level.

A - WRONG; The AOP does not direct placing the other S/G MFRV in manual because it will operate as designed with a loss of VA-10. Plausible; The examinee may believe that because MFP speed control is in manual, #2 MFRV must be put in manual to prevent level control instabilities (auto level control fighting manual feed pump operation).

B - WRONG; Due to the course operation of the local valve handwheel, control of S/G level using only local operation of the MFRV is extremely difficult and not the suggested action of the procedure.
Plausible; The examinee may believe this to be the preferred action because the AOP directs the MFRV be placed on "Local-Manual" control in the step following the instruction to place the SGFP speed controls in manual and auto pump control would dampen out course valve operation.

C - WRONG; Manually adjusting only one SGFP would cause instabilities in the system as the other SGFP automatically tries to react to changes.
Plausible; The examinee may believe because the control power is vital, the loss is facility dependent for both components and therefore requires both be placed in manual.

References

AOP-2504C, R3C7; Pg. 7, St. 3.5; Actions to control S/G level with loss of VA-10.

Comments and Question Modification History

01/05/11; Modified choices 'A' & 'C' to add "(C05)" and choices 'B' & 'D' to say "at" #1 MFRV vs. "of" #1 MFRV, based on Sandy Doboe's review.

01/31/11; Modified choices "A" & "D" from **"control of both S/G Feed Pump speeds"** to **"control of S/G Feed Pump speed"**, per RO validation. - rlc.

NRC K/A System/E/A **System** 057 Loss of Vital AC Electrical Instrument Bus

Number AA1.03 **RO** 3.6* **SRO** 3.6 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the Loss of Vital AC Instrument Bus: Feedwater pump speed to control pressure and level in S/G

Question #: 16 | Question ID: 1100022 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 1 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant is presently at 1% power with a startup in progress and all plant components functioning as designed.
Then, Non-Vital DC bus D-11 Main Feeder breaker malfunctions and trips, de-energizing D-11. The crew has entered AOP-2507A, Loss of D-11, investigating the cause of the loss, and is closely monitoring plant parameters for other required actions.

Which of the following actions would be required based on the loss of D-11 and the postulated condition associated with the stated action?

☑ **A** If "A" traveling screen delta-P exceeds 30", the "A" Circ Water Pump speed must be lowered or the breaker locally tripped to prevent damage to the screen because the "A" Circ Water Pump can not trip on high delta-P.

☐ **B** If a major fault occurs on bus 24A, the 24A-to-24C tie breaker must be locally tripped to prevent damage to bus 24C, because the 24A-to-24C tie breaker will not automatically trip due to a fault on bus 24A.

☐ **C** If forcing pressurizer sprays, the controlling pressure controller setpoint must be restored to normal to prevent a loss of RCS pressure due to the automatic tripping of the backup heater breakers.

☐ **D** If "A" TBCCW pump amps were to begin rising due to a failed bearing, the "A" TBCCW pump breaker must be locally tripped to prevent damage to the motor or associated wiring.

Justification

A - CORRECT; 30" is the setpoint for the automatic trip of a Circ. Pump on high screen delta-P. Although the control circuit for the trip may still have power, the "A" Circ. Pump breaker does not. Therefore, it must be tripped locally due to the loss of DC control power, or pump speed must be lowered, to prevent damage to the associated traveling screen.

B - WRONG; A fault on 24A could still trip the 24C-24A tie breaker because it is located on 24C and receives control power from DV-10. Plausible; Correct if DV-10 were lost. 24A receives control power from D-11 and none of its breakers can trip on any fault.

C - WRONG; PZR B/U Heater control or bus power will not be lost due solely to the loss of D-11.
Plausible; The loss of Non-Vital AC control power could cause this.

D - WRONG; The pump can be tripped from C06 because 480 VAC pumps supply their own control power from the AC supply.
Plausible; Examinee may believe that the loss of control power to the source bus will impact breaker control on the load bus.

References

1. AOP-2507A, R1C5, Pg. 3 of 18, St. 2 and Caution before it.
2. AOP-2590E-049, R0C1, A-9 "Circ Water Pump A Overload/Trip", Alarms with Trips, first bullet.
3. AOP-2517, R0C7, Pg. 26 of 29, Note before Step 9.1.

Comments and Question Modification History

01/05/11; Modified all four choices to simplify wording, based on Sandy Doboe's review.

NRC K/A System/E/A **System** 058 Loss of DC Power

Number AK3.02 **RO** 4.0 **SRO** 4.2 **CFR Link** (CFR 41.5,41.10 / 45.6 / 45.1)

Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Actions contained in EOP for loss of dc power

Question #: 17 | Question ID: 5000024 | ☑ RO ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 2 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

Unit 2 is operating at 100% power, steady state, BOL, with all switches and controls in their normal, steady state positions.
Then the crew notices the following parameter changes:

- steam generator pressure is lower
- main generator megawatt output is lower
- indicated feedwater temperature is lower
- reactor NI and delta-T power is higher
- there has been NO operator action.

Which one of the following events could have resulted in these abnormal conditions and what is the applicable procedure for addressing the problem?

☐ **A** Condenser backpressure rise (degraded vacuum), address with ARP-2590E (A-37), "COND VACUUM LO"

☐ **B** Sensor input to throttle pressure limiter failed (0 psig), address with ARP-2590D (DA-22), "10% TURBINE LOAD DECREASE"

☐ **C** Feedwater heater extraction steam isolation valve closed (heater 1B), address with ARP-2590D (AA-18), "HEATER 1A LEVEL HI"

☑ **D** Atmospheric Dump Valve opened (50% open), address with ARP-2590D (B-6), "ATMOSPHERIC DUMP VALVE NOT CLOSED"

Justification

CHOICE (A) - NO
WRONG: Degraded vacuum with no movement of control valves or control rods will result in no observable change in steam generator pressure. Steam flow will remain constant. Efficiency of the turbine will decrease. Turbine will perform less work. The additional energy rejected to the condenser will be removed by circulating water system. Feedwater temperature will be unchanged and reactor power will be unchanged.
VALID DISTRACTOR: Increasing backpressure will cause main generator output to decrease.

CHOICE (B) - NO
WRONG: Throttle pressure limiter is maintained in OFF during power operations to prevent undesirable load transients.
VALID DISTRACTOR: If on, the throttle pressure limiter would act to reduce turbine load.

CHOICE (C) - NO
'WRONG: Main turbine output will increase slightly with the isolation of an extraction line as extraction steam is redirected through subsequent turbine stages.
VALID DISTRACTOR: Loss of extraction will result in lower feedwater temperature.

CHOICE (D) - YES
Fully open ARV passes steam flow equivalent to approximately 7.5% reactor power. Steam flow will increase. Steam pressure will drop. With lower steam pressure, the main turbine output will drop. Feed flow will increase to maintain steam generator level. Increased feed flow with same extraction heating steam flow will result in lower feedwater temperature. The increased total steam flow will reduce average coolant temperature. The moderator temperature coefficient of reactivity will raise reactor power until equilibrium conditions are re-established. Reactor power and core delta-T will be higher, but Tave, Th and Tc will be lower.

References

1. MSS-00-C, "Main Steam System" Lesson, Revision 6, Section 19.b (Pg 27)
2. OP-2204, "Load Changes", Revision 19 (6/29/04), Attachment 6, "Temperature vs. Power Program"(Pg 42 of 46)
3. Source: INPO Bank - Q# 23848 - Used at Susquehanna 1, 08/01/2002

Comments and Question Modification History

02/01/11; Changed **"reactor coolant hot leg temperature is lower"** to **"reactor NI and delta-T power is higher"**, because the higher steam demand would raise reactor power, not lower Thot. Modified first sentence slightly to have plant at **"100% power"** and **"all switches and controls are in their normal, steady state positions."** The original "90%" power level confused validators because to test the Main Turbine Control Valves, power is lowered to 90% and the Throttle Pressure Limiter is placed in service. - rlc

NRC K/A System/E/A **System** 2.4 Emergency Procedure /Plan

Generic K/A Selected

NRC K/A Generic **System** 2.4 Emergency Procedures /Plan

Question #: **17** Question ID: **5000024** ☑ RO ☐ SRO ☐ Student Handout? ☐ Lower Order?

Rev. **2** ☑ Selected for Exam Origin: **Bank** ☐ Past NRC Exam?

Number 2.4.47 **RO** 4.2 **SRO** 4.2 **CFR Link** (CFR: 41.10,43.5 / 45.12)

Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Question #: 18 | Question ID: 1100010 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

Unit 2 is operating at 100% power with all equipment functioning normally. The grid suddenly experiences a partial loss of load resulting in the following conditions:

- System Frequency rises from 60 Hertz to 60.3 Hertz
- System voltage (on the monitored line) rises from 362 kVolts to 365 kVolts

Which of the following is the expected response of the Main Turbine and Generator?

☐ **A** The Control Valves will open to speed up the Main Turbine and allow Main Generator frequency to match grid frequency.
Main Generator load will automatically rise due to the rise in Main Generator frequency.
The voltage regulator will cause Main Generator voltage to rise to match grid voltage.
Main Generator reactive load will remain the same due to matching voltages.

☐ **B** The Control Valves will open to speed up the Main Turbine and allow Main Generator frequency to match grid frequency.
Main Generator load will automatically rise due to the rise in Main Generator frequency.
The voltage regulator will maintain Main Generator voltage relatively constant.
Main Generator reactive load will lower and possibly become leading due to the change in grid voltage.

☑ **C** Load Limit will prevent the Control Valves from opening; however, Main Generator frequency will rise to match grid frequency.
Main Generator load will remain relatively constant as a result of the Control Valves NOT moving.
The voltage regulator will maintain Main Generator voltage relatively constant.
Main Generator reactive load will lower and possibly become leading due to the change in grid voltage.

☐ **D** Load Limit will prevent the Control Valves from opening; however, Main Generator frequency will rise to match grid frequency.
Main Generator load will remain relatively constant as a result of the Control Valves NOT moving.
The voltage regulator will cause Main Generator voltage to rise to match grid voltage.
Main Generator reactive load will remain the same due to matching voltages.

Justification

C is correct. Main Generator frequency is a function of system frequency when the Main Generator is tied to the grid. With Load Limit in service, the Control Valves cannot open. If the Control valves don't move, Main Generator output will remain relatively constant. The automatic voltage regulator will maintain generator output relatively constant regardless of grid voltage. As grid voltage goes up, it more closely matches Main Generator voltage causing reactive load to lower. If grid voltage lowers enough, it may result in reactive load becoming leading.

A is incorrect. The Control Valves will NOT open to allow Main Generator frequency to match grid frequency. Generator frequency will match grid frequency with the output beaker closed. Main Generator load will NOT change as long as the Control Valves do NOT move. The function of the Main Generator voltage regulator is to maintain Main Generator output voltage relatively constant. If Main Generator output voltage is held relatively constant and grid voltage changes, then reactive load must change.
Plausible: If the examinee knows that generator output frequency stays locked in with grid frequency, then he/she may mistakenly believe that Main Turbine speed must change; therefore the Control Valves must open to raise Turbine speed and Generator frequency. If the Control Valves go open, then Generator load will increase. The examinee may believe that the Main Generator automatic voltage regulator maintains generator voltage approximately equal to grid voltage; therefore a change in grid voltage would cause an equivalent change in generator output voltage. If this were the case, then reactive load would remain relatively constant.

B is incorrect. See distractor A for explanation as to why the basis for Main Generator frequency is <u>incorrect</u>. See correct answer C for explanation as to why the basis for voltage and reactive load is <u>correct</u>.
Plausible: See distractor A for plausibility for Main Generator frequency. (Voltage and reactive load portion of the distractor is correct.)

D is incorrect. See correct answer C for explanation for as to why the basis for Main Generator frequency is <u>correct</u>. See distractor A for explanation as to why the basis for voltage and reactive load is <u>incorrect</u>.
Plausible: See distractor A for plausibility for Voltage and reactive load. (Main Generator frequency portion of the distractor is correct.)

References

MTC-00-C, R5, Pg. 39 of 79.

Comments and Question Modification History

2/03/10; Chip Griffin: Possible excessive wording in choices.

Question #: **18** | **Question ID: 1100010** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

Rev. 0 | ☑ **Selected for Exam** | **Origin: New** | ☐ **Past NRC Exam?**

NRC K/A System/E/A **System** 077 Generator Voltage and Electric Grid Disturbances

Number AK1.02 **RO** 3.3 **SRO** 3.4 **CFR Link** (CFR: 41.4, 41.5, 41.7, 41.10 / 45.8)

Knowledge of the operational implications of the following concepts as they apply to Generator Voltage and Electric Grid Disturbances: Over-excitation

Question #: 19 | Question ID: 1100011 | ☑ RO ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 2 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

While operating at 100% power, the CEA Partial Movement surveillance is being performed.
CEA # 15, a Group 6 CEA, is being exercised when it suddenly slips 100 steps and is now only 80 steps withdrawn.
The crew performs all necessary actions to stabilize the plant and are at the step for recovery of the dropped CEA.

Which of the following actions does the CEDS circuit require, in order to recover the dropped CEA?

☐ **A** CEA #15 must be selected on the CEAPDS Backup Scanner.

☐ **B** The Pulse Counter for CEA #15 must be reset on the PPC.

☑ **C** The CEA Motion Inhibit for CEA Group 6 must be bypassed.

☐ **D** Any Dropped Rod indications on RPS/NIS must be cleared.

Justification

C is correct. CEA #15 slipped enough to cause a CEA Motion Inhibit (CMI) on "Group Deviation". Therefore, the CMI must be bypassed to allow movement of the dropped CEA. Due to the design of the CMI Bypass circuit, in order to bypass the CMI for CEA #15, it must be bypassed for all of Group 6.

A is incorrect. The Backup Scanner has no input to the CEDS interlocks. The step to press and hold the CEA MOTION INHIBIT BYPASS is missing. If this button is NOT pressed and held, the dropped CEA will NOT move. The GROUP SELECTION does NOT need to be held.
Plausible: Selecting the dropped CEA on the Backup Scanner is required by procedure for monitoring the affected CEA. The examinee may believe that the scanner has input to the interlocks because it can be used to meet the Tech. Spec. requirement for monitoring of CEA misaligned, which is part of the Tech. Spec. that covers the requirement for CEA interlocks on misalignment.

B is incorrect. Although this impacts the PPC interlocks for Group 6, it has no effect on the CEA #15 individual interlocks.
Plausible: The examinee may believe that this is required because the PPC interlock for Group 6 withdrawal would be armed at this time, preventing Group 6 from being withdrawn.

D is incorrect. The Dropped CEA indication on the RPS/NI channels does not input to the applicable CEA interlock caused by the dropped CEA.
Plausible: RPS/NI channels do input to the CEA Withdrawal Prohibit interlock, but it is highly unlikely that this would be triggered with the appropriate operator actions taken when CEA #15 slipped. The examinee may believe that the probable Dropped CEA indication on the RPS/NI channels caused by the dropped CEA would effect withdrawal of CEA#15 based on the known RPS link to the CEA Withdrawal Prohibit interlock.

References

AOP-2556, R16C10, Pg. 14 of 55, Step 4.21 and 4.24.

Comments and Question Modification History

02/01/11; In the stem, changed **"describes switch manipulations that are required"** to **"actions does the CEDS circuit require"**, due to confusion over procedure required actions or circuit required actions. - rlc.
02/02/11; Per validation, removed **"based on the CEDS interlocks that were triggered by the dropped CEA"** from the stem as not all choices describe interlocks. Also reworded all four choices to make grammatically correct. - rlc.

NRC K/A System/E/A **System** 003 Dropped Control Rod

Number AA1.03 **RO** 3.6 **SRO** 3.3 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the Dropped Control Rod: Rod control switches

Question #: 20 | Question ID: 55614 | ☑ RO ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 7 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

The plant has just tripped from 100% power.

Which of the following would indicate that a Shutdown CEA insert <u>only 90 steps</u>?

☑ **A** No indicating light is energized on the Core Mimic and CEAPDS indicates 90 steps.

☐ **B** Blue indicating light is energized on the Core Mimic and CEAPDS indicates 90 steps.

☐ **C** Red indicating light is energized on the Core Mimic and pulse counting indicates 180 steps.

☐ **D** White indicating light is energized on the Core Mimic and pulse counting indicates 180 steps.

Justification

A - CORRECT; The Shutdown CEAs do not have a white indicating light on the core mimic like the Regulating CEAs have and the PPC would not detect that the CEA is not still at the top. The PPC indication is only reset if the CEA triggers the "Dropped Rod" reed switch located at zero steps withdrawn (fully inserted).

B - WRONG; In order for the PPC to energize the blue light on the core mimic, the CEA must trigger the "Rod Dropped" reed switch at zero steps withdrawn (fully inserted).
Plausible; CEAPDS would indicate 90 steps under these conditions and the blue light is energized when shutdown CEAs are normally withdraw or inserted.

C - WRONG; The CEA has partially inserted, therefore, the "red" light would be out.
Plausible; Pulse counting is correct because it does not reset until the rod bottom light reed switch is triggered.

D - WRONG; The Shutdown CEAs do not have a "white" indicating light, but a blue one instead.
Plausible; This would be correct for a Regulating CEA as the white light indicates that a Regulating CEA is neither on the bottom NOR on the top.

References

CED-01-C, R4, Pg. 30, "Reed Switch Position Indication" (Figure 23).

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 005 Inoperable/Stuck Control Rod

Number AK2.03 **RO** 3.1* **SRO** 3.3* **CFR Link** (CFR 41.7 / 45.7)

Knowledge of the interrelations between the Inoperable / Stuck Control Rod and the following: Metroscope

Question #: 21 | Question ID: 1100013 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The first step of AOP 2504B, Loss of Non-Vital Instrument Bus VR-21, requires the crew to secure Charging and Letdown.

Which of the following describes why?

☐ **A** Letdown isolates on a high temperature signal on the Letdown line. With Letdown isolated, Charging must be secured to prevent raising Pressurizer level above the Tech Spec limit.

☐ **B** With Channel Y selected, Letdown flow goes to minimum and all available Charging Pumps start. Charging must be secured to limit thermal shock to the Non-Regenerative Heat Exchanger.

☐ **C** Letdown flow goes to minimum and all pressurizer heaters are lost when Channel Y level input fails low. Letdown and Charging are secured to prevent a level insurge with heaters unavailable.

☑ **D** RB-402, Letdown Heat Exchanger Temperature Control Valve fails closed. Charging and Letdown are secured to prevent flashing downstream of the Letdown Heat Exchanger.

Justification

D is correct. The Letdown Heat Exchanger Outlet Temperature Transmitter fails causing RB-402, Letdown Heat Exchanger Temperature Control Valve to fail closed. Letdown temperature downstream of the Letdown Heat Exchanger will be much higher than normal. The pressure downstream of the Backpressure Control Valve will sufficiently low to possible cause Letdown to flash.

A is incorrect. The loss of VR-21 does not cause Letdown to isolate.
Plausible: A loss of VR-11 causes Letdown to isolate on a high temperature signal on the Letdown line. The effects of a loss of VR-21 may be confused with the effects of a loss of VR-11.

B is incorrect. Letdown flow will NOT go to minimum and all available Charging Pumps will not start. Channel Y will not be affected until the backup power supply battery dies.
Plausible: If the examinee believes that Channel Y (the normally controlling channel) fails high on a loss of VR-21, then Letdown will go minimum and all available Charging Pumps will start.

C is incorrect. Channel Y level is powered by VA-20, and, therefore, will not fail low on loss of VR-21.
Plausible: Channel Y will fail low if VA-20 had been lost, which would cause Letdown flow to go to minimum and all available Charging Pumps to start. With the PZR heaters failed, the control system will be unable to ensure the PZR water stays saturated on the level rise, resulting in inadequate pressure control when level is later reduced to normal.

References

AOP-2504B, R3C15, Pg. 6 of 49, Step 3.1 and Note before it.

Comments and Question Modification History

02/01/11; Modified the following due to validator input:
- Changed stem and applicable choices from "**isolate** Charging and Letdown" to "**secure** Charging and Letdown" as charging is not isolated for a loss of VR-21.
- Changed choice "C" from "**maximum due to the loss of Channel Y.**" to "**minimum due to Channel Y level input failing low.**" to improve validity of the distracter. Modified the Justification accordingly.
- Removed the word "**high**" from the answer justification. - rlc.

02/02/11; Per validation, expanded question from just asking "**Why**?". - rlc.

NRC K/A System/E/A **System** 028 Pressurizer (PZR) Level Control Malfunction

Number AK3.05 **RO** 3.7 **SRO** 4.1 **CFR Link** (CFR 41.5,41.10 / 45.6 / 45.13)

Knowledge of the reasons for the following responses as they apply to the Pressurizer Level Control Malfunctions: Actions contained in EOP for PZR level malfunction

Question #: 22 | Question ID: 1000047 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 2 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

A Radwaste Discharge of the "A" CWMT has just been started with an initial tank level of 87% and a maximum authorized discharge flow rate of 100 gpm.
When the tank has been discharging for exactly 16 minutes, the following indications exist:
"A" CWMT level at 82%.
Discharge flow recorder (FR-9050) is indicating approximately 72 gpm.
Discharge flow integrator (FI-9050) indicates approximately 1150 gallons have been discharged.

Assuming tank level indications are accurate and there are 320 gal / % level, which of the following actions should be taken?

☐ **A** Readjust the discharge flow control valve to raise the discharge rate based on the flow recorder.

☐ **B** Readjust the discharge flow control valve to lower the discharge rate based on the flow integrator.

☑ **C** Secure the discharge, then recommence by controlling the discharge flow rate based on tank level change.

☐ **D** Secure the discharge, flush the sample line for five minutes, then recommence the discharge at a lower flow rate.

Justification

C: CORRECT; The flow instrument must be considered inop, 2617A directs securing the discharge and recommencing using delta-level method.

A: WRONG; Based on change in tank level, discharge flow rate is too high.
Plausible: examinees may chose this distractor if they believe actual flow is too low based on FR-9050 reading.

B: WRONG; Based on change in tank level, discharge flow rate is too high; however, the actual flow rate must be determined to continue the discharge
Plausible: examinees may chose this distractor if they determine the actual flow rate is too high and don't realize that actual flow rate must be determined in order to continue the discharge.

D: WRONG; flushing the sample lines will NOT necessarily resolve the flow rate discrepancy.
Plausible: examinees may chose this distractor if they feel that a partially clogged sample line is the reason for the flow rate discrepancy.

References

1. SP-2617A, R29C6, Precaution 3.4
2. SP-2617A, R29C6, Attachment 3

Comments and Question Modification History

02/01/11; TYPO - Changed flow integrator number from "**(FR-9050)**" to "**(FI-9050)**". - rlc.

NRC K/A System/E/A **System** 059 Accidental Liquid Radwaste Release

Generic K/A Selected

NRC K/A Generic **System** 2.1 Conduct of Operations

Number 2.1.23 **RO** 4.3 **SRO** 4.4 **CFR Link** (CFR: 41.10 / 43.5 / 45.2 / 45.6)
Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Question #: 23 | Question ID: 1000046 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 1 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

If a fire in the plant causes the 25' 6" cable vault spreading room deluge to activate, the Fire procedure AOP 2559 directs the fire brigade to wedge open the 25' 6" cable vault spreading room East door to stairway 10 (back stairway to the Control Room), and the door from the bottom of stairway 10 to the outside.

What is the reason for this action?

☐ **A** Allows unobstructed access for fire hoses to be brought into the area from the hose station located by the Aux. Building access point.

☑ **B** Prevents deluge water from over-flowing into the DC switchgear rooms by allowing it to flow outside.

☐ **C** Provides a ventilation flow path from the outside to help purge smoke from the affected fire area.

☐ **D** Ensures access to and from the fire area in the event that the fire disables the keycard readers.

Justification

B - CORRECT; ventilation passages between the cable spreading room and the DC switchgear rooms are equipped with 3" high coffer dams, providing the stairwell as a drain path ensures that the dams are not over-flowed.

A - WRONG; The deluge should be more than adequate; but, if hoses are required, they are available in the area.
Plausible; The doors would have to be open if the fire brigade needed to use the Aux. Building access point hose station.

C - WRONG; This type of action would be evaluated and initiated by the fire brigade, not proceduralized.
Plausible; Opening the doors would create a "chimney" effect by allowing a draft from the outside to the upper level cable area.

D - WRONG: Only the bottom stairwell door has a reader and all doors can be overridden using keys.
Plausible; A fire in this area could possibly disable the security locks and not all personnel have security keys.

References

AOP-2559, R8, 1.2 - Discussion section, second paragraph.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 067 Plant fire on site

Number AK3.04 **RO** 3.3 **SRO** 4.1 **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: Actions contained in EOP for plant fire on site

Question #: 24 | Question ID: 1183154 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 1 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The plant is shut down and beginning a refueling outage, with the following conditions:

- Shutdown Cooling has just been placed in service.
- All RCPs have been secured.
- RCS Tcold = 275°F.
- RCS Pressure = 235 psia.
- All plant systems and components are configured normally for the existing mode of operation.

Then, a pipe break in the RCS occurs, resulting in a Small-Break LOCA inside containment.

How is the mitigation strategy for a LOCA different in the existing mode, as compared to Mode 3 or higher?

☑ **A** Due to the existing system and component alignments required for SDC operation, Safety System components will not automatically start or align to ensure RCS Inventory Control and Core Heat Removal.

☐ **B** Due to the existing system and component alignments required for SDC operation, Safety System components that do automatically start or align to mitigate the accident will result in over pressurizing the SDC piping system.

☐ **C** Due to the existing mode required blocking of ESAS actuation, Safety System components will not automatically start or align to ensure RCS Inventory Control and Core Heat Removal.

☐ **D** Due to the existing mode required blocking of ESAS actuation, Safety System components that do automatically start or align to mitigate the accident will result in over pressurizing the SDC piping system.

Justification

A - Correct; The procedural guidance for a LOCA while in Mode 4 or below, is contained in OP-2207, Plant Cooldown, Att. 9, Step G, Actions for a LOCA. The HPSI pumps must be taken out of PTL and the safety injection systems must be re-aligned, to allow safety injection flow to occur.

B - Wrong; This would occur if HPSI were maintained fully operable, however, OP-2207 requires the HPSI pumps be placed in P-T-L. Plausible: If the HPSI pumps were not inoperable, they would start and possibly over pressurize the SDC system. Examinee (RO) may not recall HPSI being inop at these parameters.

C - Wrong; The CTMT High Pressure SIAS actuation can not be blocked and manual safety system valves that are not remotely manipulated have been re-aligned in this mode.
Plausible; Low RCS pressure SIAS and CIAS is blocked in this mode and would not automatically actuate. Also, if the LOCA were to occur at the end of the outage, there would not be enough decay heat to pressurize CTMT enough to trigger the a SIAS.

D - Wrong; The HPSI pumps could easily raise RCS pressure above the SDC isolation valve interlock setpoint, preventing the valves from being opened if they were closed. However, this interlock has been permanently altered to prevent it from closing the isolation valves on a high system pressure.
Plausible; The examinee may believe the interlock on the SDC system isolation valves is only blocked from closing the valves due to the present mode of operation due to the danger of inadvertent system isolation on a failed signal.

References

OP-2207, R28C5, Attachment 9, Step G

Comments and Question Modification History

02/01/11; Revised stem and distracters based on validation feedback. - rlc.

NRC K/A System/E/A **System** 074 Inadequate Core Cooling

Generic K/A Selected

NRC K/A Generic **System** 2.4 Emergency Procedures /Plan

Number 2.4.9 **RO** 3.8 **SRO** 4.2 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Question #: N/A | Question ID: 8310054 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☑ Past NRC Exam?

PARENT

A plant cooldown is in progress using OP 2207.
The protected facility is Facility 1.
RCS temperature is 270° F and pressure is 375 psia with the 'A' & 'B' RCPs running.
The Shutdown Cooling System is in recirc for the warmup/pressurization leak check using the 'B' LPSI pump.
The 'A' HPSI pump is energized and in "Pull-To-Lock".
'B' and 'C' HPSI pump breakers are racked out.
Both HPSI header stop valves are closed from C-01.

Then, a seismic event occurs resulting in the following plant conditions:
- Pressurizer level and pressure are dropping rapidly.
- Containment pressure is 4.5 psig and rising.
- Both running RCPs have been secured.

Which of the following describes actions that must be taken for this event?

☑ **A** Take the 'A' HPSI Pump handswitch out of "Pull-To-Lock" and verify it automatically starts on SIAS.
Manually open the HPSI header stop valves on C-01.
Secure 'B' LPSI pump and realign LPSI pumps for Safety Injection.

☐ **B** Take the 'A' HPSI Pump handswitch out of "Pull-To-Lock" and verify it automatically starts on SIAS.
Manually open the HPSI header stop valves on C-01.
Verify 'A' LPSI pump automatically starts and aligns for safety injection.

☐ **C** Override and open the HPSI Injection valves and manually start the 'A' HPSI pump.
Verify the HPSI header stop valves automatically open on C-01.
Secure 'B' LPSI pump and realign LPSI pumps for Safety Injection.

☐ **D** Override and open the HPSI Injection valves and manually start the 'A' HPSI pump.
Verify the HPSI header stop valves automatically open on C-01.
Verify 'A' LPSI pump automatically starts and aligns for safety injection.

Justification

A - Correct; The Emergency Response Guidance for a LOCA while in Mode 4 or below, is contained in OP-2207, Plant Cooldown, Att. 9, Step 8, Actions for a LOCA while shutdown. The High CTMT Pressure SIAS can NOT be blocked. Therefore, the 'A' HPSI just needs to be taken out of PTL to start injecting into the RCS. Also, the LPSI alignment prevents injection until realigned, these actions are directed by OP 2207 in response to a LOCA.
B - Wrong; This would occur if Facility 1 LPSI were maintained fully operable. However, this is not possible if the SDC system is aligned for warmup/pressurization leak check.
C - Wrong; The HPSI injection valves are NOT overridden closed to prevent HPSI injection into the RCS at low pressure. The HPSI injection header is "key-locked" closed from C01and will NOT open on an ESAS signal.
D - Wrong; The HPSI injection valves are NOT closed, and the LPSI system is NOT properly aligned to inject into the RCS, in this mode.

References

OP-2207, Attachment 9, Step 8, Actions for a LOCA while shutdown

Comments and Question Modification History

Reworded stem and distracters per Exam Reviewer's feedback. 02/17/09

Clarified reference nomenclature in question justification per NRC comments. 02/19/09

NRC K/A System/E/A **System** 2.4 Emergency Procedure /Plan

Generic K/A Selected

NRC K/A Generic **System** 2.4 Emergency Procedures /Plan

Number 2.4.9 **RO** 3.8 **SRO** 4.2 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Question #: 25 | Question ID: 1161069 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The plant was operating at 100% power when parts from an RCP impeller broke free, circulated into the core region and caused several fuel pins to leak.
The damaged RCP then trips, resulting in a plant trip.
All systems and components responded normally on the trip.

The RO is now checking radiation monitors on the Plant Process Computer (PPC) in the performance of EOP-2525.

Which one of the following PPC radiation monitor indications would be reading abnormally high based on the existing conditions?

☐ **A** Containment Atmosphere radiation monitors.

☐ **B** Steam Generator Tube Leak N-16 monitors.

☑ **C** Auxiliary Building Area radiation monitors.

☐ **D** Main Steam Line radiation monitors.

Justification

C - CORRECT: The fuel failure would show up on Rad. Monitors in the Aux Building due to the continued RCP Bleedoff flow to the VCT.

A - WRONG: In addition to the leaking fuel pins, an RCS leak inside Containment would be necessary to cause a noticeable rise in the CTMT High Range monitors. The fuel remains in the RCS which will provide significant shielding for the CTMT High Range RMs.
Plausible; Due to the significant rise in RCS radiation levels from fuel failure, the examinee may think that fuel failure is enough to cause a rise in the Containment High Range radiation monitors.

B - WRONG: Because the N-16 rad monitors are calibrated to be sensitive to that specific isotope, a leak in a SG tube or the RCS would be required for them to see it.
Plausible; The N-16 rad monitors are very sensitive and will respond to fission product radiation at a high enough level. The examinee may feel that fuel pin leakage is enough to cause the N-16 RMs to see a rise.

D - WRONG: The MSL Rad. Monitor only see the main steam header, or "shine" from CTMT, which should not exist based on the given conditions.
Plausible; With leaking fuel pins, the examinee may feel that the Main Steam Line RMs are sensitive enough to see the "shine" from the RCS.

References

RCS-00-C, R8C1, Pg 39 of 117, 11.b. RCP, Design and Operating Characteristics, paragraph aligned to Figure 18 & 19.

Comments and Question Modification History

02/01/11; Changed "A" from CTMT **High Range** rad. monitor to CTMT **Atmosphere** rad. monitor. - rlc

NRC K/A System/E/A **System** 076 High Reactor Coolant Activity

Generic K/A Selected

NRC K/A Generic **System** 2.1 Conduct of Operations

Number 2.1.19 **RO** 3.9 **SRO** 3.8 **CFR Link** (CFR: 45.12)

Ability to use plant computers to evaluate system or component status.

Question #: N/A | Question ID: 6100069 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☑ Past NRC Exam?

PARENT

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

NO Comments or Question Modification History at this time.

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

Question #: 26 | Question ID: 1154362 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

A plant startup is in progress with the reactor presently at 3% power. While making preparations for transitioning to Mode 1, an Excess Steam Demand occurs in containment and the reactor is tripped. On the trip, the RO observes the following plant conditions:

- Four (4) CEAs are stuck fully withdrawn.
- Bus 24 A & 24C are de-energized due to an electrical fault on 24C.
- Bus 24E is de-energized (aligned to 24C).
- All other applicable buses are energized.
- "B" Charging Pump handswitch in Pull-To-Lock and aligned to 22E.
- "C" Charging Pump indicates running by handswitch lights.
- Charging Flow indicates ten (10) gpm on C02.
- Aux. Building PEO reports indication of discharge relief lifting on the "C" Charging Pump.
- Pressurizer Level is 10%, lowering.
- Pressurizer Pressure is 1700 psia, lowering.

Which of the following procedures should the RO utilize to mitigate the stuck CEAs?

☐ **A** AOP 2558, Emergency Boration via Alternate Charging Path.

☑ **B** EOP 2541, Appendix 3A, Commencing Emergency Boration.

☐ **C** EOP 2541, Appendix 23, Restoring Power to 24C/24E from Unit 3.

☐ **D** EOP 2540 (2540A), Functional Recovery (Reactivity Control).

Justification

B - CORRECT: EOP-2525 contains Contingency Actions to recover reactivity control. These actions must first be tried before moving on to other procedures.

A - WRONG; Use of actions within this AOP to mitigate this casualty is not permitted at this time.
Plausible; This AOP is the original source of the actions to combat this casualty, which have been integrated into the Contingency Actions.

C - WRONG: Use of this EOP action to mitigate this failure is not permitted at this time.
Plausible; This would be a correct choice if reactivity control became an issue after transitioning to a subsequent EOP.

D - WRONG: Use of this EOP action to mitigate this failure is not required as of yet.
Plausible; This would be the correct choice if the Contingency Actions to establish reactivity control failed.

References

1. OP-2260, "Unit 2 EOP User's Guide", R9C2; EOP-2525 Implementation Guide, 1.b. second and third bullets.
2. EOP-2525, R23, Pg. 3 of 26, Step 1.c Contingency Action "c.1".
3. EOP-2541, Standard Appendix, Appendix 3, R0C0, Emergency Boration, Step 1.

Comments and Question Modification History

2/03/10; Chip Griffin: Correct answer modified to "EOP-2541, Appendix 3".

NRC K/A System/E/A **System** A11 RCS Overcooling

Number AA2.1 **RO** 2.9 **SRO** 3.3 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the (RCS Overcooling): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

All Exam Questions Designated RO or SRO (Includes "Parents" + "Originals")

Question #: N/A | Question ID: 54362 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 2 | ☐ Selected for Exam | Origin: Parent | ☑ Past NRC Exam?

PARENT

While operating at 100% power, a plant trip occurs. While carrying out EOP-2525, Standard Post Trip Actions, the operators observe the following plant conditions:

* All CEAs are inserted.
* All buses are energized.
* Pressurizer Level is 10%, lowering.
* Pressurizer Pressure is 1700 psia, lowering.
* Tavg is 505 °F, lowering.
* RCS subcooling is 100 °F, rising.
* Feeding both SGs with Main Feedwater.
* #1 SG level 15% and dropping.
* #2 SG level 42% and rising.
* #1 SG pressure 450psia and dropping.
* #2 SG pressure 650 psia and dropping.
* Containment pressure 1.5 psig, rising.
* SJAE rad monitor activity rising.
* #2 MSL rad monitor activity rising.
* No rad monitors in alarm.

Which procedure will the operators implement next?

☐ **A** EOP 2532, Loss of Coolant Accident

☐ **B** EOP 2534, S/G Tube Rupture

☐ **C** EOP 2536, Excess Steam Demand

☑ **D** EOP 2540, Functional Recovery

Justification

SRO ONLY QUESTION - Samples 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

CHOICE (A) - NO
WRONG: Multiple events (SGTR and ESDE) require entry into the functional recovery procedure.
VALID DISTRACTOR: Pressurizer pressure is lowering, pressurizer level is lowering.

CHOICE(B) - NO
WRONG: Multiple events (SGTR and ESDE) require entry into the functional recovery procedure.
VALID DISTRACTOR: Pressurizer pressure is dropping, no containment rad monitor alarms.

CHOICE (C) - NO
WRONG: Multiple events (SGTR and ESDE) require entry into the functional recovery procedure.
VALID DISTRACTOR: #1 S/G level is dropping.

CHOICE (D) - YES
Multiple events are in progress (SGTR and ESDE with failure of MSI), requiring entry into the functional recovery procedure.

References

1. OP-2260, "Unit 2 SOP User's Guide", Revision 1 (7/11/02) (Pg 9,10 of 32)
2. EOP-2541, Appendix 1, "Diagnostic Flowchart", Revision 000 (10/2/03) (Pg 1 of 1)

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** A11 RCS Overcooling

Number AA2.1 **RO** 2.9 **SRO** 3.3 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the (RCS Overcooling) Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Question #: 27 Question ID: 56043 ☑ RO ☐ SRO ☐ Student Handout? ☑ Lower Order?
Rev. 0 ☑ Selected for Exam Origin: Bank ☐ Past NRC Exam?

A ten (10) gpm leak has developed in the Charging Line, upstream of the Regenerative Heat Exchanger.

Which of the following indications would result approximately one minute after the leak starts?

☑ **A** Regenerative Heat Exchanger Outlet Temperature (TI-221) INCREASES.

☐ **B** Letdown Heat Exchanger Outlet Temperature (TI-224) INCREASES.

☐ **C** Charging Header Pressure (PI-212) DECREASES.

☐ **D** Charging Line Temperature (TI-229) DECREASES.

Justification

A - CORRECT; Letdown flow through the Regen HX transfers heat to Charging. If the heat transfer rate remains the same, then a lower Charging flow would result in a higher charging differential temperature through the HX. The Charging inlet temperature remains the same; therefore, Charging outlet temperature must be higher.

B - INCORRECT; Letdown HX outlet temperature remains the same. RBCCW flow is automatically controlled to maintain a set Letdown temperature.
Plausible: The examinee may think that lower Charging flow through the Regen HX results in higher Letdown temperature.

C - INCORRECT; Charging pressure will remain essentially the same unless the leak is extremely large.
Plausible: The examinee may think that a leak in the Charging line would cause a loss of system pressure.

D - INCORRECT; Charging line temperature would rise with lower charging flow through the Regen HX and constant letdown flow.
Plausible: The examinee may think that lower flow would mean less heat input from the Regen HX.

References

CVCS One-Line Drawing; Figure 2B

Comments and Question Modification History

12/03/10; Chip Griffin: Question is very difficult without prints, due to need to know instrument locations.
Disagree, question solicits system response based on knowledge of normal system flow path and general thermodynamic principals.
However, "10 gpm" was added to the stem to quantify the charging pipe failure as a leak and not a rupture.

02/01/11 - Per validator feedback, restructured stem in to **two sentences** and added the time qualifier of **"approximately one minute after the leak starts"**.

NRC K/A System/E/A **System** A16 Excess RCS Leakage

Number AK2.1 **RO** 3.2 **SRO** 3.5 **CFR Link** (CFR: 41.7 / 45.7)

Knowledge of the interrelations between the (Excess RCS Leakage) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question #: 28 | Question ID: 1100015 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant is operating at 100% power, steady state when both 6.9 kV buses are de-energized due to an internal fault on the NSST.

Which of the following RPS trips, although no longer credited in the Accident Analysis, is designed to activate first on this failure and trip the reactor?

☑ **A** RCP Underspeed, when any two RCPs begin to slow down.

☐ **B** RCS Low Flow, when any of the RCPs begin to slow down.

☐ **C** High Power, when delta-T begins to rise on low RCS flow.

☐ **D** TM/LP, when RCS Tcold begins to rise on low RCS flow.

Justification

A - CORRECT; Although credit is no longer taken for the RCP Underspeed trip, it will immediately sense any speed drop in the RCPs and trip the reactor before RCS flow has a chance to degrade to its trip setpoint.

B - WRONG; Due to the RCP fly wheels, RCS flow will be relatively slow in dropping to its trip setpoint in this event, compared to RCP Underspeed.
Plausible; As credit is no longer taken (removed from Tech. Specs.) for the RCP Underspeed trip, it may be considered non-operational.

C - WRONG; RCS flow must first drop for Thot to rise and increase Delta-T Power.
Plausible; Thot will rise rapidly on the loss of all 4 RCPs, causing a rapid rise in Delta-T power. This would instantly translate to a rapid rise in Q-Power, which is about 5% below the High Power trip setpoint.

D - WRONG; RCS Tcold will not rise until Natural Circulation begins, long after the reactor has tripped.
Plausible; Tcold, which will go up on the loss of forced flow, is a direct input to the Thermal Margin/Low Pressure (TM/LP) calculation and has a very large impact on the trip setpoint (14.28 psi/°F).

References

1. RPS-01-C, R6, Pg. 8, b. Setpoint Bases 3) and Pg. 17, b. Design and Operating Characteristics, 1) RCP Underspeed.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 003 Reactor Coolant Pump System (RCPS)

Number K5.02 **RO** 2.8 **SRO** 3.2 **CFR Link** (CFR: 41.5 / 45.7)

Knowledge of the operational implications of the following concepts as they apply to the RCPS: Effects of RCP coastdown on RCS parameters

Question #: 29 | Question ID: 1176391 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

While operating at 100% power with all systems aligned normally.
Then, numerous alarms for Reactor Coolant Pump (RCP) "A" were received.
Containment sump level suddenly begins rising faster than before the RCP alarms.
Containment atmosphere radiation monitors begin to rise.

The following data was obtained for the "A" RCP:

TIME:	0000	0800	1600
RCS Pressure	2250 psia	2250 psia	2250 psia
MIDDLE SEAL	1050 psig	760 psig	720 psig
UPPER SEAL	545 psig	250 psig	210 psig
VAPOR SEAL	60 psig	30 psig	15 psig

Based on this data, how many RCP seals are considered either failed or just degraded by 1600 hours?

☐ **A** Seals Failing = 2
Seals Degraded = 0

☑ **B** Seals Failing = 2
Seals Degraded = 1

☐ **C** Seals Failing = 1
Seals Degraded = 2

☐ **D** Seals Failing = 1
Seals Degraded = 1

Justification

At 1600, the following D/P's exist: Lower = 1515 psid, Middle = 510 psid, Upper = 190 psid
B - CORRECT; The Middle seal has a delta-P greater than 200 psid (not failed), but less than 650 psid and is, therefore, degraded. The upper Seal has a D/P of less than 200 psid and is, therefore, considered failed.

A - WRONG; Two seals have NOT failed because the Middle seal D/P is >200 psid.
Plausible; Examinee may consider the Middle seal as failed because it's D/P is < 600 psid.

C - WRONG; The Lower seal is NOT degraded with the given D/P.
Plausible; Examinee may consider Lower seal degraded because its D/P is over 1500 psid.

D - WRONG; Only one seal has failed, the Upper (<200 psid).
Plausible; Examinee may consider Lower seal failed because its D/P is over 1500 psid.

References

OP-2201C, R18C9, Pg. 54, St. 4.15, RCP Seal Failure Determination w/o the PPC.

Comments and Question Modification History

02/01/11; Per validation feedback, added CTMT sump suddenly begins to rise and CTMT rads going up. Also add Vapor Seal pressures. Answer becomes 2 failed seals (Upper and Vapor) and one degraded (Middle). - rlc.

NRC K/A System/E/A **System** 003 Reactor Coolant Pump System (RCPS)

Number A3.03 **RO** 3.2 **SRO** 3.1 **CFR Link** (CFR: 41.7 / 45.5)

Ability to monitor automatic operation of the RCPS, including: Seal D/P

Question #: N/A | Question ID: 76391 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 3 | ☐ Selected for Exam | Origin: Parent | ☐ Past NRC Exam?

PARENT

While operating at 100% power the following data for "A" Reactor Coolant Pump (RCP) was obtained in response to alarms:

TIME:	0000	0800	2400
VAPOR SEAL	60 psig	60 psig	60 psig
UPPER SEAL	545 psig	250 psig	250 psig
MIDDLE SEAL	1390 psig	1240 psig	790 psig
RCS Pressure	2250 psia	2250 psia	2250 psia

Based on this data, which RCP seals have failed or degraded and what action should be performed?

☑ **A** The upper seal has failed and the middle seal has degraded. A controlled plant shutdown should be performed.

☐ **B** The upper seal has failed and the middle seal has failed. The plant should be tripped.

☐ **C** The middle seal has failed and the lower seal has degraded. A controlled plant shutdown should be performed.

☐ **D** The middle seal has failed and the lower seal has failed. The plant should be tripped.

Justification

OP-2301C Attachment 4
At 0000, the following delta-P's exist:
Lower=845
Middle=845
Upper=485
The upper Seal has a delta-P of less than 485# and is not considered failed per 2301C r16 c4. The ARP requires monitoring of seal DP.
By 0800 the middle upper

References

NO Comments or Question Modification History at this time.

NRC K/A System/E/A System

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

Question #: 30 | Question ID: 1100016 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant is operating at 100% power with all systems and components aligned normally.

Then, VCT level transmitter, LC-227 fails low (<8%), triggering a VCT Lo-Lo Level alarm.
All other plant systems and components respond as designed.

What automatic function(s) would occur as a result of this instrument failure and what action(s) would be required?

☐ **A** Charging pump suction from the VCT isolates causing all running charging pumps to trip on low suction.
Ensure letdown is secured and verify local VCT level matches level indication on C-02.

☐ **B** If the VCT is aligned for Auto Make-Up, the Auto Make-Up valve (2-CH-512) would open and blend to the VCT.
Ensure reactor power does NOT change with the automatic blended make-up just started to the VCT.

☐ **C** Charging Pumps will automatically be aligned to take a suction on the Boric Acid Tanks.
Secure charging and letdown flow and monitor reactor power for changes due to boration.

☑ **D** VCT Outlet to the charging pumps closes and the Charging Pumps suction is aligned to the RWST.
Secure Charging and Letdown flow and monitor reactor power for changes due to boration.

Justification

D is correct; at 8% decreasing, LC227 sends a signal to close the VCT outlet valve, 2-CH-501, and opens the suction flow path to the RWST, 2-CH-504 (normally open) and 2-CH-192.

A is wrong; The VCT outlet isolation (Charging Pump suction) will close; however, the Charging Pump suction from the RWST will open. With any Charging Pump flow path, the Charging Pumps will not trip on low suction pressure.
Plausible: The examinee may believe that the charging Pump suction pressure trip is prevented due to the higher pressure from the VCT than the RWST.

B is wrong; CH-515 will NOT automatically open on a VCT lo-lo level.
Plausible: CH-515 will automatically open when the VCT reaches the low level auto makeup setpoint; however, this is from a different level transmitter. The examinee may not realize that these two functions are controlled by two different level transmitters.

C is wrong; CH-514, The Charging Pump suction from the Boric Acid Tanks will NOT open on a lo-lo level in the VCT.
Plausible: The examinee believe that it's logical for the Charging Pumps to take a suction form the Boric Acid Storage Tanks on a lo-lo level in the VCT. The Boric Acid Tanks are aligned to the Charging Pump suctions during a normal blended makeup; therefore, it would be logical to assume the suction flow path is the same for a lo-lo level in the VCT.

References

ARP-2590B-027, R0C1, C-7, "VCT Level Lo Lo".

NO Comments or Question Modification History at this time.

NRC K/A System/E/A | **System** 004 | Chemical and Volume Control System

Generic K/A Selected

NRC K/A Generic | **System** 2.1 | Conduct of Operations

Number 2.1.7 | **RO** 4.4 | **SRO** 4.7 | **CFR Link** (CFR: 41.5 / 43.5 / 45.12 / 45.13)

Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation.

Question #: 31 | Question ID: 1155557 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

During the performance of OP 2207, Plant Cooldown, for refueling, the crew has just opened Shutdown Cooling Suction Isolation Valves, 2-SI-651 and 2-SI-652, with concurrent operation of the "A" and "B" RCPs.

The following conditions are noted:
- o Pressurizer level is 55% and lowering.
- o RCS temperature is 235°F and stable.
- o RCS pressure is 245 psia and lowering.
- o Quench Tank level is 50% and stable.
- o Primary Drain Tank (PDT) level is 20% and stable.
- o RWST level is 97% and slowly rising.
- o RBCCW Surge Tank level is 50% and stable.

Which of the following is the cause of the above indications and what action is procedurally directed?

☐ **A** Shutdown Cooling Excess Purification Isolation Valve, 2-CH-024, was inadvertently opened prior to the initiation of Shutdown Cooling.
Start all available Safety Injection Pumps as required to maintain Pressurizer level 35 to 70% and dispatch a PEO to close Shutdown Cooling Return to Letdown System Isolation, 2-CH-024.

☐ **B** Shutdown Cooling Suction Relief Valve (Between 2-SI-651 and 2-SI-652), 2-SI-469, is either leaking past its seat or is partially open.
Start all available HPSI Pumps as required to maintain Pressurizer level 35 to 70% and immediately close Shutdown Cooling (inside CTMT) Suction Isolation Valve, 2-SI-652.

☑ **C** LPSI/CS Test Header Isolation valve, 2-SI-460, is either leaking past it's seat or was NOT fully closed prior to the initiation of Shutdown Cooling.
Start available Charging Pumps and HPSI Pumps as required to maintain Pressurizer level 35 to 70% and dispatch a PEO to close SDC Recirculation Isolation Valve, 2-SI-460.

☐ **D** One of the two SI Pump Minimum Flow Recirculation Valves, 2-SI-659 or 2-SI-660, was inadvertently opened or is leaking past its seat.
Start all available Charging Pumps as required to maintain Pressurizer level 35 to 70% and immediately close LPSI Pump Minimum Flow Recirculation Valve, 2-SI-659 or 2-SI-660.

Justification

C is correct. SDC Recirculation Isolation Valve, 2-SI-460, is a direct path to the RWST and is listed in OP 2207, Plant Cooldown, Attachment 11, as one of the SDC leakage flow paths. When the SDC System is pressurized by opening Shutdown Cooling Suction Isolation Valves, 2-SI-651 and 2-SI-652, Pressurizer level will begin to lower with 2-SI 460 partially open or leaking past its seat.

A is incorrect. Shutdown Cooling Excess Purification Isolation, 2-CH-024, is NOT a direct path to the RWST.
Plausible: Shutdown Cooling Return to Letdown System Isolation, 2-CH-024, is listed in OP 2207, Plant Cooldown, Attachment 11, as one of the SDC leakage flow paths. If the examinee is NOT familiar with the SDC connections to external systems, then he/she may believe that this is a flow path to the RWST.

B is incorrect. Shutdown Cooling Suction Relief Valve (Between 2-SI-651 and 2-SI-652), 2-SI-469, is NOT a flow path back to the RWST.
Plausible: Shutdown Cooling Suction Relief Valve (Between 2-SI-651 and 2-SI-652), 2-SI-469, is listed in OP 2207, Plant Cooldown, Attachment 11, as one of the SDC leakage flow paths. If the examinee is NOT familiar with the SDC connections to external systems, then he/she may believe that this is a flow path to the RWST.

D is incorrect. 2-SI-659 and 660 are always open to provide minimum flow protection for the HPSI Pumps. Each LPSI Pump Minimum Flow Isolation Valve is manually isolated (locally) to ensure there is no flow back to the RWST.
Plausible: If the examinee is unfamiliar with the minimum low alignment required for for Shutdown Cooling operations. Additionally, 2-SI-659 and 660, along with the individual LPSI Minimum Flow Recirculation Valves, are listed listed in OP 2207, Plant Cooldown, Attachment 11, as one of the SDC leakage flow paths.

References

1. OP 2207, Attachment 9, Step G, Plant Cooldown Conditional Actions
2. OP 2207, Attachment 11, Step I3, Potential Leak Paths While on SDC.

Comments and Question Modification History

02/01/11; Per validation, corrected SI-659 & -660 name in choice "D" ("**LPSI**" changed to "**SI**") and 2-SI-460 name in choice "C" - rlc

NRC K/A System/E/A System 005 Residual Heat Removal System (RHRS)

Question #: **31** | **Question ID:** 1155557 | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

Rev. 0 | ☑ **Selected for Exam** | **Origin:** Mod | ☐ **Past NRC Exam?**

Number A2.04 **RO** 2.9 **SRO** 2.9 **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 RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: RHR valve malfunction

Question #: N/A | Question ID: 55557 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☐ Past NRC Exam?

PARENT

The plant is on Shutdown Cooling using the "B" LPSI pump.

The following conditions exist:
- o RCS level is at the Reactor Vessel Flange and slowly decreasing
- o RCS temperature is 100 degrees F.
- o RCS pressure is 15 psia.|o Quench Tank level is slowly increasing.
- o PDT level is steady.|o RWST level is slowly increasing.
- o RBCCW Surge Tank level is steady.

Which of the following is the most likely leakage path causing the RCS level decrease?

☐ **A** RCS cold leg drain valve leakage.

☐ **B** PORV leakage.

☑ **C** SDC recirculation to RWST (2-SI-460) leakage.

☐ **D** 'B' SDC Heat Exchanger tube leakage.

Justification

AOP 2572, Figure 5.2. Note: Question also correlates to RO Objective 7264

References

NO Comments or Question Modification History at this time.

NRC K/A System/E/A System

Number	RO	SRO	CFR Link

Question #: **32** | **Question ID: 1150052** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

Rev. 0 | ☑ **Selected for Exam** | **Origin: Mod** | ☐ **Past NRC Exam?**

The plant has experienced a reactor trip due to a large break LOCA and the following conditions exist one hour after the event:

- All safety related equipment is operating as expected.
- RWST level is 40% and lowering.
- ESAS actuation cabinet 5 has just been lost due to blown fuses and restoration will take several hours.

Which of the following is a required operator action when the SRAS (Sump Recirculation Actuation Signal) setpoint is reached, based on the above conditions?

☐ **A** The "A" LPSI pump must be manually secured from C-01 and RBCCW to Spent Fuel Pool Cooling, 2-RB-8.1A must be manually closed locally.

☑ **B** The "A" LPSI pump must be manually secured from C01 and RBCCW flow must be manually established through the "A" SDC Heat Exchanger.

☐ **C** RBCCW flow must be locally established through the "A" SDC Heat Exchanger and SI-659, minimum flow isolation, must be manually closed locally.

☐ **D** Spent Fuel Pool Cooling, 2-RB-8.1A must be manually closed locally and SI-659, minimum flow isolation, must be manually closed locally.

Justification

B is Correct; With the loss of VA-10, ESAS Actuation Cabinet #5 will be de-energized and SRAS will NOT be generated on Facility 1. Because the SRAS is "facility dependent", and only Facility 2 SRAS has power, only the Fac. 2 equipment will respond when the SRAS setpoint is reached.

A is wrong; 2-RB-8.1A is already closed due to the SIAS signal.
Plausible: The highest heat load on RBCCW is seen during a SRAS and 2-RB-8.1A is closed to ensure adequate heat sink for the CAR coolers during the Design Base Accident.

C is wrong; It is not necessary to close SI-659 because it is in series with SI-660, which will be closed by Facility 2 ESAS.
Plausible: Both SI-659 and SI-660 fail to the open position, so they would require action on a total loss of signal.

D is wrong; RB-8.1A would have closed on the SIAS before VA-10 was lost.
Plausible: RB-8.1A must be closed to ensure adequate heat sink capacity of RBCCW during a SRAS.

References

1. AOP-2504C, R3C8, Pg. 3, St. 1.2 - Discussion, last bullet.
2. EOP-2532, R29C1, Pg. 39, St. 48

Comments and Question Modification History

02/01/11; Per validation, choice "B", changed **"locally"** to **"manually"**. Also, changed cause of ESAS actuation cabinet failure from **"VA-10 power loss"** to **"blown cabinet fuses"** - rlc

NRC K/A System/E/A **System** 006 Emergency Core Cooling System (ECCS)

Number A4.05 **RO** 3.9 **SRO** 3.8 **CFR Link** (CFR: 41.7 / 45.5 to 45.8)

Ability to manually operate and/or monitor in the control room: Transfer of ECCS flowpaths prior to recirculation

Question #: N/A | Question ID: 5000052 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☑ Past NRC Exam?

PARENT

An automatic reactor trip and SIAS have occurred while operating at power.

30 minutes into the event, the following conditions exist:
- Crew is performing EOP-2532 "Loss of Coolant Accident"
- CET temp 320°F
- pressurizer pressure 28 psia
- containment pressure 9 psig
- SIAS, CIAS, EBFAS, MSI and CSAS have actuated
- all equipment is functioning per design

After several RWST LEVEL CH LO/LO alarms are received, the operator determines that SRAS has NOT actuated. Choose the NEXT correct action in response to the SRAS failure.

☐ **A** Manually stop both Charging pumps.

☐ **B** Manually close both Gravity Feed valves (CH-508/509).

☑ **C** Manually stop both Low Pressure Safety Injection pumps.

☐ **D** Manually close both RWST header outlet valves (CS-13.1A/B).

Justification

SRO ONLY QUESTION - Samples 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

CHOICE (A) - NO
WRONG: Charging pumps stopped after ensuring automatic actions of SRAS
VALID DISTRACTOR: Charging pumps are stopped after SRAS

CHOICE (B) - NO
WRONG: Gravity feed valves are closed after ensuring automatic actions of SRAS
VALID DISTRACTOR: Gravity feed valves are closed after SRAS

CHOICE (C) - YES
Directed in Step 47 of EOP-2532.

CHOICE (D) - NO
WRONG: RWST outlet valves closed after ensuring automatic actions of SRAS
VALID DISTRACTOR: RWST outlet valves closed after SRAS

References

1. EOP-2532, "Loss of Coolant Accident", Revision 23 (3/31/04) (Pg 39, 40 of 95)

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 026 Containment Spray System (CSS)

Number A2.02 **RO** 4.2* **SRO** 4.4* **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 CSS; and (b) based on those predictions, use procedures to correct, control,or mitigate the consequences of those malfunctions or operations: Failure of automatic recirculation transfer

Question #: 33 | Question ID: 1100048 | ☑ RO | ☐ SRO | ☑ Student Handout? | ☐ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

While operating at 100% power, it was determined that PORV, RC-402, was leaking by causing Quench Tank level to rise to the high level alarm. PORV Isolation valve RC-403 was closed and the appropriate action statement was entered. In accordance with the Annunciator Response Procedure, .

- The Quench Tank was drained to the PDT to restore level to 50%.
- PDT level has risen 5%.
- RCS temperature remains constant.
- NO boric Acid or PMW has been added to the system.
- Assume there is NO other leakage from any other component.

What affect will this event have on a 4 hour, manual leak rate calculation?

☐ **A** The identified leak rate will be the same and the unidentified leak rate will be lower.

☐ **B** Both the identified leak rate and the unidentified leak rate will be lower.

☐ **C** Both the identified leak rate and the unidentified leak rate will be higher.

☑ **D** The identified leak rate will be lower and the unidentified leak rate will be the same.

Justification

D is correct. The The leakage from the PORV results in a loss of inventory from the RCS; however, the Pressurizer level Control system maintains Pressurizer level at 65%. The inventory comes from the VCT. The Unidentified Leak Rate compares the loss of VCT level with the additional PDT level and determines that no inventory was lost or unaccounted for. The Identified Leak Rate sees a rise in PDT level as an addition to the system and calculates a negative (less positive) leak rate.

A is incorrect. The Identified Leak Rate calculation sees the rise in PDT level as an addition of inventory; therefore, Identified Leak Rate cannot remain the same (it must be lower).
Plausible: The examinee may see a rise in PDT level as a loss of RCS inventory; therefore, Identified Leak Rate must be higher (not the same).

B is incorrect. The Unidentified leak Rate accounts for the change in VCT level as well as the change in PDT level. The calculation sees NO change in total system volume; therefore, Unidentified Leak Rate does NOT change.
Plausible: The examinee may think that both leakage calculations see a rise in PDT level as an increase in inventory.

C is incorrect. The Identified Leak Rate sees a rise in PDT level as an increase in RCS inventory; therefore, the Identified Leak Rate is negative.. The Unindentifed Leak Rate assumes that the inventory change in the VCT is equal to the inventory rise in the PDT; therefore, there is NO inventory lost from the system.
Plausible: The examinee may think that both leak rate calculations see a loss of RCS inventory; therefore, the leak rate calculations are higher.

References **Provided**

SP 2602A, Manual Leak Rate Determination

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 007 Pressurizer Relief Tank/Quench Tank System (PRTS)

Number K1.03 **RO** 3.0 **SRO** 3.2 **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 PRTS and the following systems: RCS

Question #: 34 | Question ID: 78985 | ☑ RO ☐ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 3 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

The plant is operating at 100% in a normal configuration with Bus 24E being supplied by Bus 24C. The "C" RBCCW Pump suddenly trips. In accordance with AOP 2564, Loss of RBCCW, the "B" RBCCW Pump is started and flow is restored to the "B" RBCCW Header.

What is the status of the following three components, assuming NO other operator actions are performed?

1. The position of hand switch "SIAS/LNP ACTUATION SIGNAL HS-6119A", on breaker A504?
2. The "RBCCW PUMP B SIAS/LNP START MANUALLY BLOCKED" annunciator?
3. The "B" RBCCW Pump, if a subsequent SIAS or LNP were to occur?

☑ **A**
1. BLOCK
2. In alarm
3. Will NOT start

☐ **B**
1. NORMAL
2. In alarm
3. Will start

☐ **C**
1. NORMAL
2. NOT in alarm
3. Will NOT start

☐ **D**
1. BLOCK
2. In alarm
3. Will start

Justification

A - CORRECT; The "SIAS/LNP ACTUATION SIGNAL HS 6119A on breaker A504 is left in the BLOCK position during normal operation with the "B" RBCCW Pump as the spare. Therefore, the "RBCCW PUMP B SIAS/LNP START MANUALLY BLOCKED" annunciator will NOT be lit until the "B" RBCCW Pump is started. When the "B" RBCCW Pump is started in place of the "C" RBCCW Pump, the annunciator will alarm. If HS 6119A is NOT repositioned to "NORMAL", then the "B" RBCCW Pump will be prevented from starting on a subsequent SIAS or LNP.

B - WRONG; The switch is not put in Normal when the pump is powered from the other Facility.
Plausible; Status if "Pull-To-Lock" (P-T-L) feature of Pump Handswitch was what prevented pump from starting (true for Facility 2).

C - WRONG; This is the status of the Handswitch for the Facility 2 power supply breaker to 24E.
Plausible; Normal status for components applicable to the other facility.

D - WRONG; In "Block", the switch is designed to prevent the pump from starting.
Plausible; Condition if switch needed to be manipulated to prevent pump from starting with pump handswitch out of P-T-L.

References

AOP-2564, R4C2; Pg. 3; "Discussion"
Pg. 16, St. 6.1

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 008 Component Cooling Water System (CCWS)

Number K2.02 **RO** 3.0* **SRO** 3.2* **CFR Link** (CFR: 41.7)

Knowledge of bus power supplies to the following: CCW pump, including emergency backup

Question #: 35 | Question ID: 1167778 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The plant is shutting down for a refuel outage with the following existing conditions:

- "A" & "B" RCPs operating.
- The "A" SDC Heat Exchanger has just been placed in service.
- The crew is presently stabilizing RCS temperature.

A leak in which of the following components would result in a loss of level in the Reactor Building Closed Cooling Water (RBCCW) Surge Tank?

☐ **A** Letdown Heat Exchanger

☐ **B** "A" SDC Heat Exchanger

☐ **C** Primary Sample Cooler

☑ **D** Blowdown QT Heat Exchanger

Justification

D is correct. The Blowdown Heat Exchanger is the only component listed where RBCCW system pressure is higher than the other liquid system pressure.

A is incorrect. RBCCW is at a lower system pressure than the Letdown System at this point; therefore a tube leak in the Letdown Heat Exchanger would cause a rise in RBCCW Surge Tank level.
Plausible: If the examinee thought that Letdown System pressure in the Letdown Heat Exchanger was at a lower system pressure than RBCCW.

B is incorrect. System pressure in the SDC HX is higher than RBCCW System pressure; therefore, a tube leak would cause RBCCW Surge Tank level to rise.
Plausible: During normal operation, RBCCW system pressure is at a higher than SDC system pressure.

C is incorrect. The Primary sample Cooler is used to cool liquid from various locations in the RCS; however, there is no mechanism to reduce the pressure of the samples.
Plausible: The examinee may equate small tubing size for RCS sample flow with low pressure.

References

1. RBC-00-C, R5, Pg. 6 of 73, System Description ("equipment served" list).
2. RBC-00-C, R5, Pg. 37 of 73, c. - "RCS In-Leakage"

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 008 Component Cooling Water System (CCWS)

Number K1.02 **RO** 3.3 **SRO** 3.4 **CFR Link** (CFR: 41.2 to 41.9 / 45.7 to 45.9)

Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems: Loads cooled by CCWS

Question #: N/A | Question ID: 67778 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☐ Past NRC Exam?

PARENT

A tube leak in which of the following components would result in contamination of the Reactor Building Closed Cooling Water (RBCCW) system?

- ☑ **A** Letdown Heat Exchanger.
- ☐ **B** Spent Fuel Pool Heat Exchanger.
- ☐ **C** PDT and Quench Tank Heat Exchanger.
- ☐ **D** RBCCW Heat Exchanger.

Justification

The Letdown Heat Exchanger is the only component listed where RBCCW system pressure is lower than the other liquid system pressure.

References

NO Comments or Question Modification History at this time.

NRC K/A System/E/A System

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

Question #: **36** | **Question ID:** 1100017 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 0 | ☑ Selected for Exam | **Origin:** New | ☐ Past NRC Exam?

Which of the following design features maintains the Pressurizer Spray lines warm?

☐ **A** A small hole is drilled into the disk of each of the spray valves.

☐ **B** The spray valves are designed to leak past their seats.

☑ **C** A small bypass valve is installed around each spray valve.

☐ **D** The spray valves have a mechanical stop to prevent full closure.

Justification

C is correct. A 3/4 inch line is installed around the Pressurizer Spray valves. A throttle valve is used to limit flow to 1-1.5 gpm. The bypass line is installed to ensure the spray lines stay warm to prevent thermal shock to the spray nozzle should the spray valves open suddenly. The bypass line also helps to maintain the Boron concentration in the Pressurizer equal to the Boron concentration in the RCS.

A is incorrect. There are no holes drilled through the spray valves seats.
Plausible: Some RCS valves have holes drilled through the seats to prevent thermal binding of the valve. Example: SDC Isolation Valve, 2-SI-652.

B is incorrect. The spray valves do not have designed leak leakage.
Plausible: Some RCS valves have design leakage past the seats to prevent thermal transients. Example: SDC Total Flow Valve, 2-SI-306.

D is incorrect. The spray valves do not have mechanical stops on the valves.
Plausible: Some system valves have mechanical stops. Example: Feedwater Heater Normal Level Control Valves have mechanical stops to prevent full closure.

References

RCS-00-C, R8C3, Pg 24 of 112, second paragraph

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 010 Pressurizer Pressure Control System (PZR PCS)

Number K4.01 **RO** 2.7 **SRO** 2.9 **CFR Link** (CFR: 41.7)

Knowledge of PZR PCS design feature(s) and/or inter-lock(s) which provide for the following: Spray valve warm-up

Question #: 37 | Question ID: 1100018 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant is at 100% power, steady state, with all systems operating as designed.

Then, RPS Channel "B" Core Protection Calculator (CPC) malfunctions such that the RCS Tcold used to calculate RPS trips and pretrips is two degrees higher than actual Tcold (i.e.; Actual Tcold = 545°F, calculated Tcold used by CPC = 547°F).

All inputs to RPS are unchanged and all other CPC circuits are functioning as designed.

Which of the following would result from this malfunction?

☐ **A** Channel "B" Q-Power output would be from the channel's Delta-T Power.

☐ **B** Channel "B" would have a Power Trip Test Interlock (PTTI) actuated.

☑ **C** Channel "B" TM/LP trip setpoint would be closer to actual RCS pressure.

☐ **D** Channel "B" output would cause a CEA Withdrawal Prohibit to actuate.

Justification

C - CORRECT; The highest of the 2 Tcolds is used to generate the LSSS setpoints. Tcold is an input to the TM/LP trip setpoint, and is derived by the function P-trip = 2215 x Qdnb + 14.28 x Tcold - 8240. Therefore, a failure in the CPC causing the Tcold used to rise 2 degrees will result in about a 29 psi rise in the TM/LP setpoint.

A - WRONG; Thot is not said to change and Tcold is NOT an input into the refinement of the NI detector input. Therefore, Tcold going up would result in drop in Delta-T power.
Plausible; As an actual change in Tcold would result in a change in NI power seen by the excore detectors, an examinee may assume the signal is compensated for Tcold. In that premise, a rise in Tcold would result in a drop in the NI calculated power, making Delta-T power the "high-select" choice.

B - WRONG; This requires a failure in the RPS Calibration and Indication Panel or NI drawer.
Plausible; The RPSCIP is just above the CPCs in the RPS channels and has controls to adjust numerous inputs into the CPCs. A failure of a CPC calculated value could imply a failure of the RPSCIP.

D - WRONG; This requires two TM/LP pretrips or High Power pretrips to activate.
Plausible; a TM/LP pretrip is one of the triggers for a CEA Withdrawal Prohibit.

References

RPS-01-C, R6C4, Pg. 22 of 80, 10) Thermal Margin / Low Pressure Trip setpoint development.

Comments and Question Modification History

02/01/11; Per validation, swapped choice "A" wording around to improve grammar (eliminate "passive tense"), removed "**RPS**" from choices "B" & "D" and changed "**pressurizer**" in choice "C" to "**RCS**". - rlc

NRC K/A System/E/A **System** 012 Reactor Protection System

Number K6.07 **RO** 2.9* **SRO** 3.2* **CFR Link** (CFR: 41.7 / 45/7)

Knowledge of the effect of a loss or malfunction of the following will have on the RPS: Core protection calculator

Question #: **38** | Question ID: **1150064** | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. **0** | ☑ Selected for Exam | Origin: **Mod** | ☐ Past NRC Exam?

Which one of the following conditions would require immediate entry into EOP-2525, "Standard Post-Trip Actions", if the condition were to occur inadvertently with the reactor operating at 100% power?

[Consider each one separately and assume all other plant systems and components function as designed.]

☐ **A** Containment Isolation Actuation Signal on both Facilities.

☑ **B** Main Steam Isolation Signal on Facility 1 only.

☐ **C** A spurious trip of the bus 24A-to-24C Tie Breaker.

☐ **D** Both Channel "A" S/G Levels fail low on #1 S/G only.

Justification

B - CORRECT; Because each facility of ESAS is designed to complete the safety function, either facility actuating will result in both MSIVs going closed, resulting in either a manual or automatic reactor trip.

A - WRONG; Spurious CIAS is addressed by AOP-2571, "Inadvertent Emergency Core Cooling System Initiation", which provides direction for maintaining power operation while addressing the problems of inadvertent isolation.
Plausible; Examinee may think that with both facilities of CIAS actuated, a plant trip is immanent or required.

C - WRONG; A spurious trip on the bus xtie will cause an LNP on 24C. The bus will de-energize and then re-energize on the "A" EDG. The reactor will NOT need to be tripped because only one MG set was lost.
Plausible; Examinee may think that a loss of a vital 4160 bus at 100% power for at least 12 seconds will require a plant trip.

D - WRONG; It takes 2 channels of Low S/G level to cause a plant trip, but they must be 2 separate channels, NOT 2 of the same channels on one S/G.
Plausible; Examinee may remember that any 2 channels of S/G level failing low will cause a plant trip, but not understand that it must be 2 separate channels.

References

1. ESAS-00-C, R3C5, Pg. 19 of 73, 10. - Main Steam Isolation (MSI).
2. ARP-2590A-105, R0, C-01, A-27 "Stm. Gen. Pres. Lo Lo"

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 013 Engineered Safety Features Actuation System (ESFAS)

Generic K/A Selected

NRC K/A Generic **System** 2.4 Emergency Procedures /Plan

Number 2.4.2 **RO** 4.5 **SRO** 4.6 **CFR Link** (CFR: 41.7 / 45.7 / 45.8)

"Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions."

Question #: N/A | Question ID: 5000064 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☑ Past NRC Exam?

PARENT

Which of the following conditions would NOT require immediate entry into EOP-2525, "Standard Post-Trip Actions", if the condition were to occur inadvertently with the reactor operating at 100% power?

☑ **A** Containment Isolation Actuation Signal on both Facilities

☐ **B** Main Steam Isolation Signal on Facility 1 only

☐ **C** Overcurrent trip of normal feeder breaker to Bus 25B from NSST

☐ **D** Loss of VA-20 with loss of HV to Linear Range CH 'D'

Justification

CHOICE (A) - YES
Spurious CIAS is addressed by AOP-2571, "Inadvertent Emergency Core Cooling System Initiation", which provides direction for maintaining power operation while addressing the problems of inadvertent isolation.

CHOICE (B) - NO
WRONG: Each facility of ESAS can complete the safety function. Both MSIVs will close resulting in either a manual or automatic reactor trip.
VALID DISTRACTOR: Applicant may think that both Facilities must actuate to trip reactor.

CHOICE (C) - NO
WRONG: An overcurrent trip on the bus normal feed will lock out the alternate feed from the RSST. Fast transfer will not occur. The bus will de-energize and its associated RCP breakers will trip. The reactor will trip on loss of low speed of two RCPs.
VALID DISTRACTOR: Applicant may think that fast transfer will occur, maintaining power to the RCPs.

CHOICE (D) - NO
WRONG: Loss of VA-20 trips RPS CH 'B'. Loss of HV to a linear range NI causes associated channel trips through the PTTI. With 2 RPS high power trips, a reactor trip will occur.
VALID DISTRACTOR: Applicant may think that loss of HV will not trip the variable high power on Channel D since the NI output will fail low.

References

1. AOP-2571, "Inadvertent Emergency Core Cooling System Initiation", Revision 4 (12/14/98) (Pg 4 of 15)
2. IHE-00-C, "In-House Electrical System" Lesson, Revision 9 (Pg 10 of 86)
3. Source: INPO Bank - Q# 16 - Used at Braidwood 1, 6/7/1999

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.4 Emergency Procedure /Plan

Generic K/A Selected

NRC K/A Generic **System** 2.4 Emergency Procedures /Plan

Number 2.4.2 **RO** 4.5 **SRO** 4.6 **CFR Link** (CFR: 41.7 / 45.7 / 45.8)

"Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions."

Question #: **39** | Question ID: 1155727 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 1 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

While operating at 100% power, the plant experiences an intersystem LOCA from the "B" RCP seal. A Low Pressurizer Pressure SIAS is automatically initiated during the performance of EOP 2525.

The crew has subsequently entered EOP 2532, Loss of Coolant Accident, and has just closed the RBCCW header isolations to containment. All plant systems and components are responding as designed.

What is the expected response of CAR Fan motor current from this point forward and why?

☐ **A** Amps will initially decrease due to the rise in Containment air temperature which lowers air density.

☐ **B** Amps will decrease due to the CAR Fans shifting to low speed at a high Containment pressure of 9 psia.

☐ **C** Amps will increase due to the increase in air flow from opening the spring-loaded discharge dampers.

☑ **D** Amps will initially increase due to higher air density caused by the increase in the moisture content.

Justification

D is correct. When the LOCA is isolated (RBCCW CTMT Isolation Valves are closed), the Intersystem Relief Valves on the RBCCW piping inside Containment will open resulting in a LOCA in Containment. This results in an increase in moisture content which causes CAR Fan loading (amps) to increase. In fact, the CAR Fans automatically swap to low speed on a SIAS to prevent overloading the CAR Fan motors during a LOCA or ESD.

A is incorrect. CAR Fan amps will NOT lower.
Plausible: Containment air temperature will rise. Warmer air is generally less dense than cold air. This alone would cause the CAR Fan motors to work less, lowering amps.

B is incorrect. The CAR Fans are already running in low speed.
Plausible: The examinee may believe the CAR Fans swap to slow speed when Containment Spray is initiated at 9 psia.

C is incorrect.
Plausible: If the examinee believes that the spring loaded discharge dampers automatically open on a LOCA or ESD in Containment. The discharge dampers will open as a result of a large pressure spike in Containment caused by a sudden large break. This event does NOT cause a large pressure spike.

References

CCS-01-C, Rev 10/1, Page 54.

Comments and Question Modification History

02/01/11; Per validation, changed stem wording from "**completed performing all steps required for LOCA isolation.**" to "**closed the RBCCW header isolations to containment.**" - rlc.

NRC K/A System/E/A **System** 022 Containment Cooling System (CCS)

Number A4.01 **RO** 3.6 **SRO** 3.6 **CFR Link** (CFR: 41.7 / 45.5 to 45.8)

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

Question #: N/A | Question ID: 55727 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 1 | ☐ Selected for Exam | Origin: Parent | ☐ Past NRC Exam?

PARENT The "D" CAR fan is operating in SLOW speed when a large break LOCA occurs inside Containment.||Which statement correctly describes how load on the "D" Containment Air Recirculation fan is affected by this event, and the reason why?||

☐ **A** Load decreases due to higher air temperature and lower air density. |

☑ **B** Load increases due to higher air density and higher air pressure. |

☐ **C** Load remains the same because fan blade pitch compensates for density and pressure changes. |

☐ **D** Load increases due to the CAR fan shifting to fast speed. |

Justification

Load increases due to higher air density and higher air pressure.$$$$$$

References

NO Comments or Question Modification History at this time.

NRC K/A System/E/A System

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

Question #: 40 | Question ID: 1141019 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The plant is at 100% power with the following equipment alignments:
"A", "B" and "C" CAR Fans are running in FAST with the Emergency RBCCW discharge valves open.
"D" CAR is secured with only the Normal RBCCW discharge valve open.
All other systems and components are aligned normally and operating as designed.

Then, the following occurs:
The plant trips due to a Small Break LOCA inside CTMT.
On the trip, 24C is de-energized due to a fault on the bus.
Facility 2 ESAS does not process any actuation signals due to a fault in Actuation Cabinet 6.
CTMT pressure is 4 psig and rising slowly.

Which of the following containment fan alignments is required for the above conditions, per the applicable EOP?

☑ **A** Shift the "B" CAR fan to SLOW and start the "D" CAR fan in SLOW.
Open the "D" CAR Fan Emergency RBCCW discharge valve.

☐ **B** Verify the "B" CAR fan is running and start the "D" CAR fan in FAST.
Open the "D" CAR Fan Emergency RBCCW discharge valve.

☐ **C** Shift the "B" CAR fan to SLOW and start the "D" CAR fan in FAST.
Close the "D" CAR Fan Normal RBCCW discharge valve.

☐ **D** Shift the "B" CAR fan to SLOW and verify the "D" CAR fan is secured.
Close the "D" CAR Fan Normal RBCCW discharge valve.

Justification

A - CORRECT; The "B" & "D" CAR fans are powered and would automatically start in Slow or shift to Slow on SIAS or UV. Also, their Emergency RBCCW discharge valve would automatically open. However, with ESAS disabled, these actions must be manually accomplished.

B - WRONG; CAR fans must be running in slow during an accident in CTMT or they will trip on Thermal Overload.
Plausible; Examinee may recognize the need to start the additional CAR fan, but not the requirement to have them running in SLOW.

C - WRONG; The CAR fans must be running at the same speed, in SLOW, or the fans and duct work will become overloaded.
Plausible; Examinee may recall for normal CTMT cooling, CAR fans must be running in FAST speed, but running both in fast during an accident might overload the ductwork (only true if 4 CAR fans are running in fast speed).

D - WRONG; Both CAR fans are required to be running during an accident in CTMT, with an open Emergency RBCCW discharge valve, because the other facility of CTMT cooling is unavailable (loss of power).
Plausible; Examinee may believe the "D" CAR fan is the "spare" or "swing" component that is not used unless the other fan on that facility is unavailable (true for normal CTMT cooling requirements).

References

1. EOP 2525, Standard Post Trip Actions, step 8.1
2. EOP 2532, Loss of Coolant Accident, Step 10.d.

Comments and Question Modification History

02/02/11; Per validation, deleted the word **"in"** from the question stem. - rlc.

NRC K/A System/E/A **System** 022 Containment Cooling System (CCS)

Number A2.03 **RO** 2.6 **SRO** 3.0 **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 CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Fan motor thermal overload/high-speed operation

Question #: N/A | Question ID: 4100019 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☐ Past NRC Exam?

PARENT

The plant is at 100% power, NOT/NOP. All equipment is in its normal alignment except that the 'A' EDG is out for PMs. The plant trips due to a SBLOCA inside CTMT. The RSST is lost due to a fault. CTMT pressure is 2 psig and rising slowly. EOP 2525 is completed including the following contingency steps. -- Ensure at least 2 CAR fans operating with RBCCW cooling -- Place all available Containment Aux. Circ. Fans to SLOW -- Start all available PIR fans What is the status of the following containment fans for the above condition?

☑ **A** 'B' & 'D' CAR fans in Slow No Containment Aux. Circ. Fan running 'B' PIR fan running.

☐ **B** 'B' & 'D' CAR fans in Fast No Containment Aux. Circ. Fan running 'A' & 'B' PIR fans running

☐ **C** 'B' & 'D' CAR fans in Slow 'B' & 'D' Containment Aux. Circ. Fans running No PIR fan running

☐ **D** 'C' & 'D' CAR fan in Fast 'A' & 'B' Containment Aux. Circ. Fan running 'B' PIR fan running

Justification

A: correct. 'B' &'D' CAR fans are powered due to the 'B' DG energizing 24D and CAR fans start in Slow or shift to Slow on SIAS or UV. The CTMT Aux Circ fans are non-vital power and deenergize with a faulted RSST. Only 'B' PIR fan, powered from B62, has power to start and doesn't auto-start on any signal but is started in 2525. B: is incorrect. CAR fans are shifted to slow in 2525 and only the 'B' PIR fan has power. C: is incorrect. The 'B' & 'D' Containment Aux. Circ Fans are non-vital powered. Those busses are deenergized when the RSST is faulted. D: is incorrect. The CAR fans are shifted to slow on a SIAS and in 2525, and Containment Aux Fans are deenergized.

Question References not yet listed.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 022 Containment Cooling System (CCS)

Number K2.01 **RO** 3.0* **SRO** 3.1 **CFR Link** (CFR:41.7)

Knowledge of power supplies to the following: Containment cooling fans

Question #: 41 | Question ID: 1100020 | ☑ RO ☐ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

A Large Beak LOCA has occurred from 100% power operation concurrent with a loss of Bus 24C. SIAS, CIAS, EBFAS, MSI, and CSAS have all automatically actuated.

- "B" Containment Spray header flow indicates 1,210 gpm.
- RBCCW flow to each operating CAR Cooler is 2,100 gpm.

What is the status of the Containment Cooling System with regard to its ability to perform its intended function?

☐ **A** The "B" Containment Spray header has more than the required design flow. With two CAR Coolers in service, Containment temperature and pressure will NOT exceeding design limits.

☐ **B** The Containment Spray System does NOT have adequate flow to establish an effective spray pattern; therefore, the Iodine concentration in the Containment atmosphere will remain high until adequate flow is established.

☐ **C** The Containment Spray System and CAR Coolers are presently providing adequate Containment cooling; however, when SRAS occurs, Containment Spray flow will NOT be adequate to maintain core cooling.

☑ **D** The "B" Containment Spray header has less than the required design flow. With only two CAR coolers in service, Containment temperature and pressure may exceed design limits.

Justification

D is correct. The minimum design Containment Spray flow is 1300 gpm. The design of the Containment Cooling System is such that two fully functioning CAR Coolers and one fully functioning Containment Spray System are necessary to prevent exceeding design Containment temperature and pressure limits.

A is incorrect. The "B" Containment Spray header has less than the design (procedural) limit of 1300 gpm. With Bus 24C deenergized, only two CAR Coolers are available. This combination of CAR Coolers and Containment Spray with less than the design flow rate does NOT guarantee that Containment temperature and pressure limits will be maintained less than design limits.
Plausible: If the examinee does NOT know the minimum Containment Spray flow limit, then one Containment Spray header and two CAR Coolers are adequate to ensure Containment temperature and pressure will NOT exceed the design limits.

B is incorrect. With a lower than minimum flow, the spray pattern is likely affected; however, Iodine scrubbing of the Containment atmosphere is NOT the overriding function of the Containment Cooling System. Lower than design flow will impact the ability of the Containment Cooling System to ensure Containment temperature and pressure remain below design limits.
Plausible: Iodine scrubbing is a function of the Containment Spray System. The examinee may feel that two CAR Coolers is adequate to provide the required Containment Cooling and that Containment Spray is necessary to reduce Containment atmosphere Iodine concentration, limiting the radioactive release to the environment.

C is incorrect. The Containment Cooling System is NOT providing adequate heat removal from Containment due to low flow in the "B" Containment Spray header, the loss of "A' Containment Spray, and the loss of two CAR Coolers.
Plausible: If Containment Spray does NOT meet the termination criteria when SRAS initiates, then core cooling may be negatively impacted. If the examinee does NOT know the minimum Containment Spray flow limit, then one Containment Spray header and two CAR Coolers are adequate to ensure Containment temperature and pressure will NOT exceed the design limits.

References

EOP 2532, Loss of Coolant Accident, step 11.b.
Tech Spec Bases for LCO 3.6.2.1, Containment Spray and Cooling Systems.

Comments and Question Modification History

02/02/11; Per validation, added a comma to "**2100**" in the stem. - rlc.

NRC K/A System/E/A **System** 026 Containment Spray System (CSS)

Number K3.01 **RO** 3.9 **SRO** 4.1 **CFR Link** (CFR: 41.7 / 45.6)

Knowledge of the effect that a loss or malfunction of the CSS will have on the following: CCS

Question #: 42 | Question ID: 1100019 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant had tripped from 100% power on low steam generator level due to the loss of a Main Feedwater Pump.
The following plant conditions now exist:

- One pressurizer safety valve has stuck full open on the trip.
- VA-20 was lost (deenergized) on the trip.
- Facility 1 SIAS, CIAS, EBFAS have been manually actuated and verified.
- ALL plant equipment responded as designed per the given conditions.
- All Steam Dump valves are presently closed.
- Containment pressure is 3.5 psig and slowly rising.
- The crew completed EOP-2525 and has just transitioned to the applicable event specific EOP.

The US has directed the BOP to perform a plant cooldown using BOTH steam generators.

Which one of the following contains actions that are required for performing the plant cooldown?

☐ **A** Due to the loss of control power to PIC-4216 and MSI actuation, override/open both MSIV Bypass Valves and open the Condenser Steam Dump valves using TIC-4165 on C05.

☐ **B** Due to the loss of control power to PIC-4216 and the ADVs, utilize the Foxboro Steam Dump Control screen on a PPC workstation to open the Turbine Bypass/Steam Dump valve.

☑ **C** Due to the loss of control power to the "B" ADV and imminent MSI actuation, open the "A" ADV using PIC-4223 on C05, and dispatch a PEO to manually operate "B" ADV locally .

☐ **D** Due to the loss of control power and MSI actuation, utilize the Foxboro Steam Dump Control screen to open both the "A" and "B" ADVs by placing each ADV controller in manual.

Justification

C - CORRECT; The "A" ADV can be opened using PIC-4223 by raising its output, but due to the loss of VA-20, the "B" ADV can only be opened locally.

A - WRONG; A containment pressure MSI cannot be overridden and the Bypass valves cannot be opened unless their opening coils are installed locally.
Plausible; Examinee may recognize that these actions are similar to those taken to cooldown during a SGTR and would be an easier way to control the cooldown rate.

B - WRONG; Although the loss of VA-20 prevents Facility 2 MSI from actuating, either facility of MSI actuating closes both MSIVs
Plausible; Examinee may remember that when the loss of VA-20 prevents a Facility 2 ESAS Actuation and deenergizes a couple steam dump controllers on the main control board. However, the Foxboro control Screen can be utilized to control one of the steam dump valves.

D - WRONG; The "B" ADV cannot be operated from the control room by any means with a loss of VA-20. The valve must be opened locally.
Plausible; The Examinee may believe that the control board controller is deenergized in a fashion similar to a momentary loss of VR-11/VR-21 and, therefore, the valves can be controlled by directly interfacing with the Foxboro Control System.

References

1. Loss-Of-Control-Power Operator Aid, R-1, on C-07
2. LP ESA-01-C, Engineered Safety Features Actuation System, Pg. 19
3. One-Line Diagram of Steam Dump/Turbine Bypass Control System, CL242

Comments and Question Modification History

12/03/10, Comment from Chip Griffin:
Added "imminent" MSI actuation to answer C. MSI does not automatically actuate until 4.43 psig in Containment.

NRC K/A System/E/A **System** 039 Main and Reheat Steam System (MRSS)

Number A2.01 **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 MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Flow paths of steam during a LOCA

Question #: **43** | **Question ID: 1150018** | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 0 | ☑ Selected for Exam | **Origin: Mod** | ☐ Past NRC Exam?

A steam generator tube rupture has occurred on #2 SG. EOP-2534, "Steam Generator Tube Rupture" has been implemented.

When isolating the #2 SG, which of the following actions is performed, in accordance with EOP-2534, to ensure a setpoint or limit is not exceeded?

☐ **A** Maintain #2 SG level below 40%, to ensure additional primary-to-secondary leakage does not over fill the SG and put water into the Main Steam lines.

☑ **B** Place #2 SG ADV in AUTO with a setpoint of 920 psia, to ensure the ADV lifts before the Main Steam Safety Valves on a potential rise in SG pressure.

☐ **C** Place #1 SG ADV in AUTO with a setpoint of 900 psia, to ensure the ADV will open and maintain the RCS Tavg below the Mode 3 limit of 532 °F.

☐ **D** Override and open the MSIV Bypass Valve on the #2 SG, to prevent the affected SG ADV from opening due to a potential rise in SG pressure.

Justification

B - CORRECT; This places the ADV in a condition to open prior to pressure in the isolated steam generator reaching the MSSV lift setpoint, and minimizing the possibility that a MSSV will open and stick in an open position.

A - WRONG; SG level is maintained ABOVE 40% to help with scrubbing of iodine entering the SG from the RCS leakage.
Plausible; The statement is true in that it would help in preventing SG level from rising high enough to spill into the Main Steam header. However, although this was a prescribed action in the past, it is not the overriding concern now.

C - WRONG; This is not a required action of EOP-2534 at this point in the event.
Plausible; This action is directed by procedure and is required under normal conditions.

D- WRONG: This is only required in EOP-2534 if the level in the affected, and isolated, SG can NOT be maintained below 90%, which would put it in danger of spilling into the Main Steam header.
Plausible; EOP-2534 does contain this action, however the existing plant status given in the stem does NOT warrant it.

References

1. EOP-2534, "Steam Generator Tube Rupture",

Comments and Question Modification History

12/03/10, Chip Griffin. Stem uses term "SG2", answers use #2 SG. Changed SG2 to #2 SG to be consistent.

02/01/11; Per validation, changed **"Once the #2 SG is isolated"** to **"When isolating the #2 SG"**. - rlc

NRC K/A System/E/A **System** 039 Main and Reheat Steam System (MRSS)

Generic K/A Selected

NRC K/A Generic **System** 2.1 Conduct of Operations

Number 2.1.32 **RO** 3.8 **SRO** 4.0 **CFR Link** (CFR: 41.10 / 43.2 / 45.12)

Ability to explain and apply system limits and precautions.

Question #: N/A | Question ID: 5000018 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☑ Past NRC Exam?

PARENT

A steam generator tube rupture has occurred on SG2. EOP-2534, "Steam Generator Tube Rupture" has been implemented. Which of the following actions is performed in accordance with EOP-2534, "Steam Generator Tube Rupture" to DIRECTLY limit the potential radiation release to the public?

☐ **A** raise ruptured SG level above 40% using TDAFW pump

☑ **B** ensuring ruptured SG ADV setpoint at 920 psia and closed

☐ **C** tripping RCPs if pressurizer press less that 1714 psia and SIAS initiated

☐ **D** entering EOP-2536 (ESDE) for a SG pressure < 800 psia and subcooling going up

Justification

CHOICE (A) - NO
WRONG: MSI blocked to facilitate controlled cooldown via preferred method (turbine bypass valves to condenser)
VALID DISTRACTOR: because MSI is blocked by procedure.

CHOICE (B) - YES
The ADV is ensured to be in auto and its setpoint is raised to a value below the upper end of the band. It is also ensured to be closed since steam pressure should be below this point. This places the ADV in a condition to open prior to pressure in the isolated steam generator reaching the MSSV lift setpoint, and minimizing the possibility that a MSSV will open and stick in an open position.

CHOICE (C) - NO
WRONG: RCP trip strategy based on worst-case LOCA concerns. Continued operation of pumps is preferable during a SGTR event to allow for a prompt controlled RCS cooldown and depressurization.
VALID DISTRACTOR: because RCP trip is directed by procedure under specified conditions.

CHOICE (D) - NO
WRONG: Functional Recovery Procedure is used to address multiple events from a symptom-based perspective. EOP-2536 should not be entered and implemented from EOP-2534 with multiple events in progress
VALID DISTRACTOR: because EOP-2534 does contain diagnosis confirmation steps and the functional recovery does address excess steam demand events.

References

1. EOP-2534, "Steam Generator Tube Rupture", Revision 22, (3/22/02), (Pg 10, 17 of 64)
2. TG-2534, Steam Generator Tube Rupture, Revision 21 (Pg 14, 22, 30, 37 of 126)
3. Source: Indian Point 3 NRC Exam, 12/2003

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.3 Radiation Control

Generic K/A Selected

NRC K/A Generic **System** 2.3 Radiation Control

Number 2.3.11 **RO** 3.8 **SRO** 4.3 **CFR Link** (CFR: 41.11 / 43.4 / 45.10)
Ability to control radiation releases.

Question #: **44** | **Question ID:** 1100021 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☑ Selected for Exam | **Origin:** New | ☐ Past NRC Exam?

The plant is in a normal configuration, operating at 100% power, when the #2 Atmospheric Dump Valves suddenly fails full open, creating a Steam Flow/Feed Flow mismatch.

Without any operator action, how will the Main Feedwater System respond to this event?

☐ **A** The steam flow detectors will sense a significant rise in steam flow. The indicated mismatch between steam flow and feed flow will generate a signal to open the #2 Main Feed Regulating Valve.

☑ **B** The steam flow detectors will not sense the increase in steam flow, but the increase will cause Steam Generator level to lower. Lowering level will cause the #2 Main Feed Regulating Valve to open.

☐ **C** The sudden decrease in Steam Generator pressures will generate a rise in Main Feed Pump speed. The Main Feed Regulating Valve differential pressures will rise resulting in the FRVs closing.

☐ **D** The steam flow detectors will sense the rise in steam flow and generate a rise in Main Feed Pump speed. The Main Feed Regulating Valve differential pressures will rise resulting in the FRVs closing.

Justification

B is correct. The steam flow detector is located downstream of the ADVs, therefore actual steam flow will be greater than indicated steam flow. Indicated steam flow and feed flow will remain nearly equal. The actual increase in steam flow will cause S/G level to lower resulting in the #2 FRV opening to attempt to restore level. The level deviation signal is NOT as strong as the steam flow/feed flow mismatch signal; therefore, actual steam generator level will eventually be maintained at a lower level than setpoint.

A is incorrect. The steam flow detectors are downstream from the ADV; therefore, they will NOT detect the rise in steam flow. As a result, the FRVs will not immediately respond to the change in steam flow.
Plausible: If the examinee does NOT realize the steam flow detectors are downstream of the ADVs then, he/she may think the system will maintain S/G levels at setpoint.

C is incorrect. As steam flow increases, Steam Generator pressures will lower and feed pump speed will rise. However, the rise in SGFP speed is due to the rise in steam flow, not the lowering of SG pressure causing a reduction in pump resistance.
Plausible: if the examinee remembers that feed pump speed rises with a rise in steam flow, but does not understand why.

D is incorrect. Indicated steam flow will NOT rise; therefore, feed pump speeds will NOT rise.
Plausible: The examinee may recognize that a rise in feed pump speed will result in a higher delta-P across the FRVs, forcing them to close slightly in order to maintain level.

References

1. MSS-00-C (Rev. 7, Change 1), Main Steam System, Page 11 of 68
2. MSS-00-C (Rev. 7, Change 1), Main Steam System, Figure 1

Comments and Question Modification History

02/02/11; Per validation, 'C' modified to be wrong and removed final outcome from all choices to eliminate speculation on complex system dynamics. - rlc.

NRC K/A System/E/A **System** 059 Main Feedwater (MFW) System

Number K4.08 **RO** 2.5 **SRO** 2.7 **CFR Link** (CFR: 41.7)

Knowledge of MFW design feature(s) and/or interlock(s) which provide for the following: Feedwater regulatory valve operation (on basis of steam flow, feed flow mismatch)

Question #: 45 | Question ID: 1154376 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The Main Steam supply line to the Turbine Driven Aux. Feedwater Pump (TDAFW) has ruptured at the "T" connection just downstream of 2-MS-4A & 4B check valves.
The plant was then tripped and the BOP was directed to close 2-MS-201 and 2-MS-202, Main Steam supply to the TDAFW pump, and isolate the steam leak.

Which of the following actions must be accomplished to allow the valves to be closed?

☑ **A** The disconnect switch for 2-MS-202 only must be closed.

☐ **B** The disconnect switches for both valves must be closed.

☐ **C** The breaker at the MCC for 2-MS-202 only must be closed.

☐ **D** The breaker closing coils for both valves must be installed.

Justification

A - CORRECT; MS-201 is normally open with its power supply and control circuit fully aligned to allow for operation from C05. MS-202 is normally open with the closing coil installed and power from B62 aligned. However, a disconnect switch is installed downstream of the breaker and left open to prevent the valve from closing if an App. 'R' fire causes a "hot-short" in the valves control circuit. Therefore, to close MS-202, its disconnect switch must first be closed. Then both valves can be operated from the C05.

B - WRONG; Only MS-202 has a disconnect switch.
Plausible; As both valves have the same "safety significance" with respect to the AFW system, both may be assumed to have a disconnect switch.

C - WRONG; The breaker is left closed to allow for position indication. Only the disconnect is left open to meet App. 'R' concerns.
Plausible; It may be remembered that the valve operator is electrically defeated, but not how. Only 480 VAC components lose their position indication when the MCC breaker is opened.

D - WRONG; The closing coils for all motor operating valves were reinstalled when the disconnect switches were installed.
Plausible; Removal of the closing coil is how the App. 'R' concern was met in the past, before a plant change that installed manual disconnect switches.

References

1. MSS-00-C (Rev. 7, Change 1), Main Steam System, Page 28 of 68, Section 20.c.

Comments and Question Modification History

Modified stem question from "accomplished to close the valves" to "accomplished to allow the valves to be closed", per feedback from Sandy Doboe. 01/06/11

NRC K/A System/E/A **System** 061 Auxiliary / Emergency Feedwater (AFW) System

Number K1.03 **RO** 3.5 **SRO** 3.9 **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 AFW and the following systems: Main steam system

Question #: N/A | Question ID: 54376 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☐ Past NRC Exam?

PARENT

The steam supply line to the Turbine Driven Aux. Feedwater Pump (TDAFW) has ruptured at the "T" connection just downstream of 2-MS-4A & 4B check valves. The plant was then tripped with the following conditions seen at the completion of EOP-2525:||- Unit 2 RSST is dead||- "B" Emergency Diesel Generator breaker will not close||- 24D is tied to 24E||NO OTHER equipment malfunctions occurred.||What action must be accomplished to close 2-MS-202 and isolate the Excess Steam Demand?||

☐ **A** Energize 24D from 14H THEN close 2-MS-202 from C05. |

☑ **B** Install the closing coil in breaker B6202, energize 24D from 14H and THEN close the valve from C05. |

☐ **C** Install the closing coil in breaker B6202, THEN close the valve from C05. |

☐ **D** Send a PEO to close the valve LOCALLY. |

Justification

MS-201 is normally open with the closing coil installed in B52. MS-202 is normally open with the closing coil removed from B62. The electrical problems will require B62 be reenergized via 14H to 24D. Local operation is not possible because the steam leak is located near 2-MS-202 (Enclosure Bldg., West side, 38-6 ft. level, near CTMT access).$$$$[Copied from Item No '3516' on 10/01/97 By RLC]

References

NO Comments or Question Modification History at this time.

NRC K/A System/E/A System

Number	RO	SRO	CFR Link

Question #: 46 | Question ID: 1179056 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The plant has just tripped from 100% power due to a loss of the grid and trip of the Main Turbine.
The following plant conditions now exist 25 seconds after the trip:

- Pressurizer pressure peaked at 2430 psia and dropped below 2395 psia after 20 seconds.
- The "A" Safety Channel NI failed at 100% at the time of the trip.
- #1 SG level = 35 % and dropping.
- #2 SG level = 25 % and dropping.
- All other plant parameters and systems are responding as designed following the trip.

How will the Auxiliary Feed Water (AFW) System respond under these conditions, to ensure plant design limits are NOT exceeded?

☐ **A** Only the Facility 1 AFW system automatically actuated 10 seconds after the trip.

☐ **B** Both Facilities of the AFW System automatically actuated 10 seconds after the trip.

☐ **C** The 3 min. and 25 sec. timer on only the Facility 1 AFW System is running and will result in automatic system actuation unless conditions change.

☑ **D** The 3 min. and 25 sec. timer on both Facilities of the AFW System is running and will result in automatic system actuation unless conditions change.

Justification

D - CORRECT; The ATWAS circuitry for 2400 psia and >20% power is NOT actuated by any safety NI channels, but by a control NI channel. AFW will still actuate on low SG level after 3 minutes and 25 seconds.

A - WRONG; The Diverse Scram System will NOT actuate due to a failure of a Safety NI Channel.
Plausible: With Safety NI Channel "A" failed high and Pressurizer pressure momentarily above 2400 psia, the examinee may believe that Facility 1 AFW is automatically actuated is affected; however, Facility 1 DSS will NOT actuate unless the Facility 1 Control NI Channel fails to >20% while Pressurizer pressure is >2400 psia.

B - WRONG; The Diverse Scram System will NOT actuate due to a failure of a Safety NI Channel.
Plausible: Since both facilities of AFW are automatically actuated on low level, the examinee may believe that the Diverse Scram System results in the actuation of both facilities of AFW.

C - WRONG; With either S/G level below the automatic AFW setpoint, both AFW facilities are actuated.
Plausible: With #2 SG level below the automatic AFW actuation setpoint of 26.8% and #1 S/G level above the setpoint, the examinee may believe that only Facility 2 AFW is automatically initiated.

References

1. AFW-00C, Auxiliary Feedwater System Lesson Text, page 5 of 54, Section 1
2. AFW-00C, Auxiliary Feedwater System Lesson Text, page 7 of 54, Paragraph j.
3. AFW-00C, Auxiliary Feedwater System Lesson Text, pages 10 and of 54.

Comments and Question Modification History

12/03/10, Chip Griffin. How long was pressure above 2400 psia?
Added that pressure was above 2395 psia for about 20 seconds.

NRC K/A System/E/A System 061 Auxiliary / Emergency Feedwater (AFW) System

Number A1.01 **RO** 3.9 **SRO** 4.2 **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 AFW controls including: S/G level

Question #: N/A | Question ID: 79056 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 1 | ☐ Selected for Exam | Origin: Parent | ☐ Past NRC Exam?

PARENT

The plant was operating normally at 100% power when it tripped due to high Pressurizer pressure (pressure spiked to 2420 psia).

- The "A" Safety Channel NI failed high at the time of the trip.
- All other plant parameters and systems responded normally following the trip.
- Pressurizer pressure lowered to 2220 psia shortly after the trip and is now stable at 2250 psia.
- Both Steam Generators are at 23% level and slowly rising.
- It has been two (2) minutes since the plant tripped.

Based on these conditions, which of the following is true?

☐ **A** The Facility 1 Aux. Feedwater System has fully actuated on high Pressurizer pressure combined with the sensed reactor power at the time of the trip.

☐ **B** Both Facilities of the Aux. Feedwater System have automatically actuated as a result of the high Pressurizer pressure trip ONLY.

☐ **C** Both Facilities of the Aux. Feedwater System have automatically actuated on high Pressurizer pressure combined with sensed reactor power at the time of the trip.

☑ **D** The Aux Feedwater System 3 min. and 25 sec. timer is running and will result in automatic system actuation unless overridden by operator intervention.

Justification

The ATWAS circuitry for 2400 psia and >10% power is not a lock in circuit. AFW will still actuate on low SG level after 3 minutes and 25 seconds.

References

NO Comments or Question Modification History at this time.

NRC K/A System/E/A System

Number	RO	SRO	CFR Link

Question #: 47 | Question ID: 1154565 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The plant is operating in MODE 5, performing Plant Heatup OP 2201, when the RSST is suddenly deenergized due to a fault.

- The "A" Diesel Generator (DG) starts, but the associated output breaker fails to automatically or manually close.
- "A" DG is emergency tripped.
- All other equipment operates as expected.
- Bus 24E is now energized from Unit 3.

Based on these conditions, which of the following statements identifies the appropriate procedure and the correct sequence of steps needed to energize Bus 24C?

☐ **A** Per EOP 2541, Appendix 23, "Restoring Electrical Power", place all four UV BUS A3 keys in "inhibit", reset the ESAS UV signal, close A305, 24C/24E Tie Breaker.

☐ **B** Per EOP 2541, Appendix 23, "Restoring Electrical Power", reset the Sequencer on Actuation Cabinet 5, reset the ESAS UV signal, close A305, 24C/24E Tie Breaker.

☑ **C** Per AOP 2502C, "Loss of Vital 4.16 kV Bus 24C", place all four UV BUS A3 keys in "inhibit", reset the ESAS UV signal, close A305, 24C/24E Tie Breaker.

☐ **D** Per AOP 2502C, "Loss of Vital 4.16 kV Bus 24C", reset the Sequencer on Actuation Cabinet 5, reset the ESAS UV signal, close A305, 24C/24E Tie Breaker.

Justification

C is correct. To allow closing A305, 24C/24E Tie Breaker, the four channels of UV for Bus 24C must be bypassed, then the UV actuation signal on Facility 1 must be reset prior to energizing Bus 24C.

A is incorrect. EOP 2541, Appendix 23 will require the same steps to be performed, however, in MODE 4 only the AOP is applicable. EOPS may only be used in MODE 3 or above.
Plausible: If the examinee feels that the EOP has better guidance or it is applicable in a lower MODE, then this procedure will work.

B is incorrect. The Sequencer on Actuation Cabinet 5 does NOT need to be reset. Additionally, the UV on Bus 24C cannot be reset unless at least three out of four undervoltage channels are bypassed.
Plausible: EOP 2541, Appendix 23, <u>and</u> AOP 2502C both require the Sequencer to be reset if it did not fire. In this case the DG started; therefore, the Sequencer fired. The examinee may feel that the Sequencer failed to actuate because the DG output breaker failed to close. Additionally, the examinee may think that the UV may be reset without bypassing all four UV channels. See above for potential for selecting EOP 2541, Appendix 23.

D is incorrect. This is the correct procedure; however, the Sequencer on Actuation Cabinet 5 does NOT need to be reset. Additionally, the UV on Bus 24C cannot be reset unless at least three out of four undervoltage channels are bypassed.
Plausible: See justification for distractor B for plausibility.

References

1. AOP 2501, Diagnostic for Loss of Electric Power, Page 3, Paragraph 1.3, Applicability
2. AOP 2501, Diagnostic for Loss of Electric Power, Page 6, Step 3.3
3. AOP 2502C, Loss of Vital 4.16 kV Bus 24C, Steps 3.36 through 3.38.

Comments and Question Modification History

02/02/11; changed Mode in stem to Mode 5 to ensure EOP-2528 is NOT applicable. - rlc.

NRC K/A System/E/A **System** 062 A.C. Electrical Distribution

Number A2.05 **RO** 2.9 **SRO** 3.3* **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 ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Methods for energizing a dead bus

Question #: N/A | Question ID: 5054565 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 1 | ☐ Selected for Exam | Origin: Parent | ☐ Past NRC Exam?

PARENT

Following a plant trip, the Emergency Diesel Generators (EDGs) are supplying their respective Bus 24C and 24D due to a failure to transfer to the Reserve Station Service Transformer (RSST).

While still in EOP-2528, "Loss of Offsite Power/Loss of Forced Circulation", the RSST is now available to supply Facility 1 electrical loads. NO electrical faults exist.

Based on these conditions, which of the following statements identifies the procedure and the CORRECT sequence of steps needed to restore Bus 24C to a normal post-trip alignment?

☑ **A** Per EOP-2541, Appendix 23, "Restoring Electrical Power", reset ESAS UV signal, parallel RSST to 24C, open D/G breaker, close Bus 24A-24C tie breaker.

☐ **B** Per AOP-2502C, "Loss of Vital 4.16 kV Bus 24C", reset ESAS UV signal, parallel RSST to 24C, open D/G breaker, close Bus 24A-24C tie breaker.

☐ **C** Per EOP-2541, Appendix 23, "Restoring Electrical Power", reset ESAS UV signal, parallel RSST to 24C, close Bus 24A-24C tie breaker, open D/G breaker.

☐ **D** Per AOP-2502C, "Loss of Vital 4.16 kV Bus 24C", reset ESAS UV signal, parallel RSST to 24C, close Bus 24A-24C tie breaker, open D/G breaker.

Justification

SRO ONLY QUESTION - Samples 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

CHOICE (A) - YES
These actions are directed in the listed sequence in Appendix 23.

CHOICE (B) - NO
WRONG: AOP-2502C does not provide step sequence for re-energizing from normal source.
VALID DISTRACTOR: AOP-2502C provides steps for energizing bus from the emergency diesel generator.

CHOICE (C) - NO
WRONG: Sequence is not correct. Tie breaker is not closed until after D/G breaker is open.
VALID DISTRACTOR: Plausible to think tie breaker should be closed before opening D/G breaker.

CHOICE (D) - NO
WRONG: AOP-2502C does not provide step sequence for re-energizing from normal source.
VALID DISTRACTOR: Plausible to think that AOP would provide specific guidance for transferring bus back to normal source.

References

1. OP-2343, "4160 Volt Electrical System", Revision 20 (9/9/04), Section 4.20 "Restoring Bus 24C to Unit 2 RSST with Emergency Diesel Generator
Supplying" (Pg 42 of 71)

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 062 A.C. Electrical Distribution

Generic K/A Selected

NRC K/A Generic **System** 2.2 Equipment Control

Number 2.2.2 **RO** 4.6 **SRO** 4.1 **CFR Link** (CFR: 41.6 / 41.7 / 45.2)

Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.

Question #: 48 | Question ID: 1100008 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant is at 30% power and lowering for a normal plant shutdown using OP 2204, Plant Shutdown. Shortly after in-house loads are transferred to the RSST, Vital DC Bus 201A suddenly deenergizes. The crew enters AOP 2505A, Loss of Vital 125 VDC Bus 201A, which requires them to dispatch an operator to locally trip the "A" Diesel Generator using the mechanical overspeed trip push button and to close both air start header isolation valves.

Why is the Diesel Generator tripped and why do the air start header isolation valves need to be closed?

☑ **A** All automatic and manual trips, with the exception of mechanical overspeed, are disabled on a loss of DC power.
The air start valves fail open on a loss of DC resulting in the Diesel continuing to roll with no protection.

☐ **B** All automatic and manual trips, with the exception of mechanical overspeed, are disabled on a loss of DC power.
Closing the air start header isolation valves will prevent the Diesel from restarting when DC power is restored.

☐ **C** The loss of vital DC control power will result in the affected Diesel starting and running with NO cooling water flow.
The air start valves fail open on a loss of DC resulting in the Diesel continuing to roll with no protection.

☐ **D** The loss of vital DC control power will result in the affected Diesel starting and running with NO cooling water flow.
Closing the Air start header isolation valves will prevent the Diesel from restarting when DC power is restored.

Justification

A is correct. The loss of Vital DC power will disable all automatic and manual trips on the "A" Diesel Generator, with the exception of the manual overspeed trip (and Fuel Rack Trip). All other trips require DC control power to actuate a trip on either the Diesel or the Generator. The air start valves are open by deenergizing a DC powered solenoid. The loss of Vital DC power will cause the associated DG to start and run on the low speed stop (920 RPM). If the manual isolation valves are not closed, the air tanks will completely depressurize.

B is incorrect. The first part is true, but the manual isolation valves are NOT closed to prevent the D/G from starting when DC power is restored. When the automatic air start solenoids are energized, they close the valves.
Plausible: If the air start isolation valves are NOT closed, then the examinee may believe that the air start solenoids require DC power to open and that a Diesel start signal is present due to an LNP signal generated at the trip.

C is incorrect. Cooling water flow to the "A" D/G is NOT affected by the loss of DC. Facility 1 Service Water is still in service (Bus 24C remains energized on the trip). Additionally, the Service Water supply valve to the "A" D/G fails open on a loss of DC power; therefore, cooling water is available at all times during this event.
Plausible: The examinee may believes that Bus 24C is lost due to a failure to fast transfer on the trip caused by the loss of DC. Although the plant did trip due to the loss of DC, and a fast transfer was NOT processed, Bus 24C remains energized from the RSST; therefore, the associated diesel does NOT have an LNP signal.

D is incorrect. See B and C above. Cooling water is NOT lost to the "A" D/G and the D/G will NOT restart when Dc power is restored.
Plausible: See B and C above. An LNP signal is NOT processed because Bus 24C remains energized . Air start header valves will NOT open when DC power is restored.

References

EDG-00-C, Page 135 of 143

Comments and Question Modification History

12/3/10, Chip Griffin. Modify the reason the Diesel starting air are closed?
Modified; "running at 900 rpm" to "continuing to roll" in choices "A" and "C".

02/02/11; **EOP**-2505A should be **AOP**-2505A. - rlc

NRC K/A System/E/A **System** 063 D.C. Electrical Distribution

Number K3.01 **RO** 3.7* **SRO** 4.1 **CFR Link** (CFR: 41.7 / 45.6)

Knowledge of the effect that a loss or malfunction of the DC electrical system will have on the following: ED/G

Question #: 49 | Question ID: 1100023 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

With the plant operating normally at 100% power, the "A" Diesel Generator has just been started for surveillance . The operator must raise frequency and voltage slightly to obtain the procedurally directed values prior to synchronizing the Diesel Generator with Bus 24C. The operator adjusts the Governor Control switch and the Auto Voltage Control Regulator switch to raise generator speed and voltage to the procedurally prescribed settings.

How will the Diesel Generator respond to the same operation of the Governor Control switch and the Auto Voltage Control Regulator switch AFTER the Diesel Generator output breaker is closed?

☑ **A** Megawatt load will rise; Reactive load will rise.

☐ **B** Diesel speed will rise; Bus voltage will rise.

☐ **C** Bus voltage will rise; Reactive load will rise.

☐ **D** Megawatt load will rise; Diesel speed will rise.

Justification

A is correct. After the output breaker is closed, raising the Governor Control switch will cause megawatt load to rise. Raising the Auto voltage Control regulator switch will cause reactive load to rise.

B is incorrect. Bus voltage will NOT rise. Bus 24C is still connected to the RSST which will determine Bus voltage. The same holds true for frequency (Diesel speed). The RSST (grid) will determine diesel generator frequency and speed.
Plausible: The examinee may believe that Bus voltage and frequency are determined by the Diesel Generator. This is true if the Diesel Generator were NOT running in parallel with the grid through the RSST.

C is incorrect. Reactive load will rise; however, bus voltage will remain constant.
Plausible; The examinee may believe that bus voltage will rise if Diesel Generator output voltage is increased. This is true if the Diesel Generator were NOT running in parallel with the grid through the RSST.

D is incorrect. Megawatt load will rise; however, Diesel speed will remain the same.
Plausible; The examinee may believe that Diesel speed will rise if the Diesel Generator governor control is taken to raise. This is true if the Diesel Generator were NOT running in parallel with the grid through the RSST.

References

OP 2346A, Rev 027-12, Section 4.5, Synchronizing and Loading "A" D/G From the Control Room.

Comments and Question Modification History

02/02/11; Reworded order of switch manipulations in question to match the order in the first paragraph, last sentence. - rlc.

NRC K/A System/E/A **System** 064 Emergency Diesel Generators (ED/G)

Number A3.13 **RO** 3.0* **SRO** 2.9 **CFR Link** (CFR: 41.7 / 45.5)

Ability to monitor automatic operation of the ED/G system, including: Rpm controller/megawatt load control (breaker-open/ breaker-closed effects)

Question #: 50 | Question ID: 1100024 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

With the plant operating normally at 100% power, the following annunciators are suddenly received:

- Main Steam Line Hi Rad/Inst. Fail, A-30 on C-01
- N-16 High, CA-19 on C-06/7
- N-16 Alert, CB-19 on C-06/7
- Process Mon Rad Hi Hi/Fail, DA-24 on C-06/7
- Process Mon Radiation Hi, DB-24 on C-06/7

Assuming the annunciators are valid for plant conditions and the PPC is NOT available, which of the following lists the Radiation Monitors that would be in alert or alarm on Radiation Monitor Panel, RC-14?

☐ **A** #1 or #2 N-16 Radiation Monitor, RM-4296A or B
A, B, or C Main Steam Line Radiation Monitor, RM-4299A, B, or C
Steam Jet Air Ejector Radiation Monitor, RM-5099

☐ **B** Steam Jet Air Ejector Radiation Monitor, RM-5099
A, B, or C Main Steam Line Radiation Monitor, RM-4299A, B, or C
Blowdown Radiation Monitor, RM-4262

☑ **C** #1 or #2 N-16 Radiation Monitor, RM-4296A or B
Steam Jet Air Ejector Radiation Monitor, RM-5099
Blowdown Radiation Monitor, RM-4262

☐ **D** #1 or #2 N-16 Radiation Monitor, RM-4296A or B
Blowdown Radiation Monitor, RM-4262
Unit 2 Stack High Range Radiation Monitor, RM-8168

Justification

C is correct. A Steam Generator Tube Rupture is the only event that would cause all of the listed annunciators to be valid. The Blowdown Radiation Monitor will be in at least an alert state based on the Process Mon Radiation Hi annunciator, on C-06/7. With the other annunciators being valid, the Steam Jet Air Ejector would be in alarm, which would generate the Process Mon Rad Hi Hi/Fail, on C-06/7. The remaining annunciators have Radiation Monitors that do NOT provide indication of an alarm or alert status on the Radiation Monitor Panel, RC-14.

A is incorrect. The Blowdown Radiation Monitor will be in at least an alert state based on the Process Mon Radiation Hi annunciator, on C-06/7.
Plausible: The examinee may believe that the Blowdown Radiation Monitor will not be in alert or alarm yet due to the inherent delay caused by its long sample line.

B is incorrect. The N-16 Radiation Monitors also provide alert and alarm indication on RC-14.
Plausible: The examinee may believe that the N-16 radiation monitors are not on RC-14 because they are generated by the PPC and provide alert or alarm indication on the PPC.

D is incorrect. The Unit 2 Stack High Range Radiation Monitor indicate an alarm of alert condition on Panel RC05E or the PPC only.
Plausible: The examinee may believe that this Radiation Monitor is on RC-14 with the vast majority of the other plant rad. monitors.

References

ARP 2590H, Rev. 005-03, Alarm Response for Control Room Radiation Monitor Panels, RC-14.

Comments and Question Modification History

12/17/10, RJA
Correct answer (C) did not include the N-16 Rad Monitors. Changed all distractors (and associated Justifications) to include an additional Rad monitor. Specifically changed C to include N-16 Rad Monitors.

02/01/11; Per validation, change "**Main Steam Radiation Monitors**" in choice "D" to "**Unit 2 Stack High Range Radiation Monitor, RM-8168**". - rlc.
02/02/11; Per validation, switch the order of the rad. Alarms in choice 'C' to the order they would actually come in. - rlc.

NRC K/A System/E/A **System** 073 Process Radiation Monitoring (PRM) System

Number A4.02 **RO** 3.7 **SRO** 3.7 **CFR Link** (CFR: 41.7 / 45.5 to 45.8)

Ability to manually operate and/or monitor in the control room: Radiation monitoring system control panel

Question #: 51 | Question ID: 73614 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

The plant is operating at 100% power with all systems and components functioning as designed.
Due to a storm that recently passed by the area at sea, marine growth is beginning to clog the cooling systems in the Intake.
The heat removal capability of the Service Water (SW) and RBCCW systems is beginning to degrade due to strainer and heat exchanger clogging.

Which of the following conditions (taken individually) would require a plant trip, per AOP-2564, Loss of RBCCW or AOP-2565, Loss of Service Water?

☐ **A** Containment temperature rises to 120°F due to rising RBCCW header temperatures with 3 CAR Fans running.

☑ **B** RBCCW Header temperatures rise above 120°F due to low Service Water flow through the heat exchangers.

☐ **C** A Service Water pump trips and flow restoration with the standby SW pump takes six (6) minutes to complete.

☐ **D** Rising RCP seal temperatures cause seal pressures to oscillate and trigger momentary seal pressure alarms.

Justification

B - CORRECT; Per AOP-2564, when RBCCW temp. >120°F due to low SW flow, the RBCCW pump must be tripped. Subsequent actions on a loss of RBCCW will require a plant trip.

A - WRONG; This would require a Tech. Spec. entry and possible a controlled shut down, but not a plant trip.
Plausible; Examinee may confuse RBCCW temp. limit with the CTMT temp. limit requiring a shut down. CTMT Tech. Spec. limit for continued operation is 120°F and exceeding it would require a plant shut down to Mode 5.

C - WRONG; Although this may lead to a plant trip, a loss of SW flow, in an of itself, does not require it.
Plausible; Examinee may be confusing the required plant trip if RBCCW flow is lost for > 5 minutes.

D - WRONG; This can occur if RCP seal problems are present, however, it does not require a plant trip.
Plausible; Examinee may recall RCP temperatures exceeding alarm limits requires a plant trip and the RCP be secured.

References

AOP-2564, R4C2; Pages 3 & 46

Comments and Question Modification History

02/01/11; Per validation, Choice "C" changed from "**greater than five (5) minutes**" to "**six (6) minutes to complete**" and corrected typo; "**past**" to "**passed**". - rlc.

NRC K/A System/E/A **System** 076 Service Water System (SWS)

Number K3.01 **RO** 3.4* **SRO** 3.6 **CFR Link** (CFR: 41.7 / 45.6)

Knowledge of the effect that a loss or malfunction of the SWS will have on the following: Closed cooling water

Question #: 52 | Question ID: 1100025 | ☑ RO ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant was operating at 100% power with the "B" Service Water (SW) pump aligned to supply Facility 2 heat loads due to planned maintenance on the "C" SW pump (pump is tagged out). All equipment is functioning as designed.

Then, the "A" RBCCW pump tripped on overload and the "B" RBCCW pump was started in its place. All applicable system and component manipulations were made in accordance with AOP-2564, Loss of RBCCW.

The following conditions now exist:

- 24E is aligned to 24D.
- "A" & "B" SW pumps are running.
- "B" SW pump is supplying Facility 2 SW loads.
- "B" & "C" RBCCW pumps are running.
- "B" RBCCW pump supplying Facility 1 RBCCW loads.
- "A" RBCCW pump and breaker are being evaluated.

Then, the plant trips due to a loss of the grid (state wide blackout).

Which of the following describes the status of the RBCCW and Service Water systems?

☑ **A** Only Facility 2 RBCCW header has flow.
Both Facilities of SW have flow.

☐ **B** Only Facility 2 RBCCW header has flow.
Only Facility 2 SW header has flow.

☐ **C** Only Facility 1 RBCCW header has flow.
Only Facility 1 SW header has flow.

☐ **D** Neither Facility of RBCCW has flow.
Only Facility 1 SW header has flow.

Justification

A - CORRECT; "B" RBCCW pump breaker switch alignment prevents pump start on an 24E ("B" EDG). However, "B" SW pump is properly aligned to Facility 2 through 24E tie to 24D. Therefore, both facilities of SW have flow.

B - WRONG; No effect on "A" SW pump ("A" EDG). Loss of RBCCW header does not impact SW.
Plausible: If examinee thinks switch alignment of "B" RB and SW pumps are linked because both are powered from 24E.

C - WRONG; "B" SW pump will not be 'selected' to trip over "B" RB pump.
Plausible; If examinee believes SIAS/LNP block switch is facility aligned to ensure like facilities (RB & SW) are started.

D - WRONG; No effect on Facility 2 SW due to "B" RB pump not starting on the "B" EDG.
Plausible; If examinee thinks "B" SW pump is lost due to loss of "B" EDG on overload (which may happen if starting 2 RB pumps simultaneously).

References

1. RBC-00C, Rev 6, RBCCW System Lesson Text, Page 17 of 73.
2. OP 2330A, Rev. 023-06, RBCCW System, Page 6 of 113.

Comments and Question Modification History

02/01/11; Per validation, added "**(pump is tagged out)**" to the stem to clarify that the "C" SW pump will NOT start. - rlc.

NRC K/A System/E/A **System** 076 Service Water System (SWS)

Number K2.04 **RO** 2.5* **SRO** 2.6* **CFR Link** (CFR: 41.7)

Knowledge of bus power supplies to the following: Reactor building closed cooling water

Question #: 53 | Question ID: 1000116 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 1 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

While operating at 100% power, the BOP notices that Instrument Air header pressure is at approximately 88 psig and slowly lowering.

Which of the following is an automatic action if Instrument Air header pressures drops below 85 psig?

☐ **A** The Instrument Air header supply to the Containment Air Receiver will automatically close and the Station Air supply to the Containment Air Receiver will automatically open.

☐ **B** The Station Air header will automatically align to supply just the Instrument Air header safety system component loads and automatically isolated from Station Air loads in Containment.

☑ **C** The Station Air header will automatically align to supply all Instrument Air header loads and the Station Air header will automatically be isolated from all normal Station Air header loads.

☐ **D** The Backup Air Supply bottles will automatically become available to supply both the Main Feed Water Regulating Valves and Main Feed Water Regulating Bypass Valves.

Justification

C - CORRECT; The pressure switch that operates 2-SA-10.1 and 2-SA-11.1 senses the pressure of the Instrument Air Receiver Tank. This is done so all of the Station Air capacity is supplied to Instrument Air if the I.A. supply to all I.A. headers is threatened.

A - WRONG; There is NO automatic swap to station air on a low Containment air pressure. This must be done manually.
Plausible; Examinee may believe that CTMT air loads would receive the "automatic" swap to SA, as CTMT entry takes a lot of time and most CTMT air loads are safety related.

B - WRONG; Station air is automatically aligned to all IA components and isolated from all SA components, not just those in CTMT.
Plausible; Examinee may believe only safety related components will be aligned due to the limited capacity of the SA compressor.

D - WRONG; Although there is a "backup air header" that can supply to the MFRVs, it is a parallel path to the normal IA supply to the valves and is always aligned. Also, there is NO automatic alignment of the backup air system to the AFRVs. This must be done manually.
Plausible; The "backup" supply to the MFRVs is designed to limit the chance of a IA header rupture causing a loss of FRV control. As it is a passive system function, the examinee may believe the AFW system, being a "safety" system, must have an automatic backup.

References

ISA-00-C, Rev. 8, Ch. 2, Station Air and Instrument Air Systems, Page 12 of 67.

Comments and Question Modification History

12/3/10, Chip Griffin, In distractor D, the phrase 'will automatically be aligned' implies that valves move or reposition. The word 'aligned' was changed to 'available' to clear up any confusion.

02/01/11; Per validation, **removed all reference to Station Air Header** status from the stem. Also, modified 'D' to remove it as a possible correct answer. - rlc

NRC K/A System/E/A **System** 078 Instrument Air System (IAS)

Number A3.01 **RO** 3.1 **SRO** 3.2 **CFR Link** (CFR: 41.7 / 45.5)

Ability to monitor automatic operation of the IAS, including: Air pressure

Question #: 54 | Question ID: 4000022 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 1 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

The plant has tripped from 100% power due to an Excess Steam Demand event inside Containment. During the performance of EOP 2525, Standard Post Trip Actions, the following conditions were noted by the BOP:

- Buses 25A and 25B are deenergized.
- Buses 24A and 24C are deenergized.
- "A" Emergency Diesel Generator (EDG) failed to automatically start. (NO faults are indicated).
- Containment pressure is 27 psig and slowly rising.
- All other plant systems and components are functioning as designed.

Which of the following describes actions that the BOP must perform per EOP 2525?

☐ **A**
1. Verify all Facility 2 safety related components have automatically started.
2. Emergency trip the "A" EDG.
3. Place the "A" RBCCW Pump in Pull-To-Lock.
4. Place the "A" SW Pump in Pull-To-Lock.

☐ **B**
1. Start the "A" EDG.
2. Verify the associated output breaker automatically closes.
3. Verify "A" SW pump automatically starts.
4. Verify "A" RBCCW pump automatically starts.

☐ **C**
1. Place the "A" RBCCW Pump in Pull-To-Lock.
2. Have the "A" RBCCW pump discharge valve throttled.
3. Start the "A" EDG and verify "A" SW pump automatically starts.
4. Manually start the "A" RBCCW Pump and have the discharge slowly opened.

☑ **D**
1. Place "A" RBCCW Pump in Pull-To-Lock.
2. Start the "A" EDG.
3. Verify the associated output breaker automatically closes.
4. Verify the "A" Service Water pump automatically starts.

Justification

D is correct. EOP 2525, states, " If Bus 24C or 24D is NOT energized and Containment pressure is greater than or equal to 20 psig, then place the associated RBCCW Pump in Pull-To-Lock, ensure the associated Diesel Generator has started, and ensure the output breaker for the associated diesel Generator is closed." With >20 psig in Containment during a LOCA or ESD and no RBCCW flow through the CAR Coolers, will cause the stagnant water to flash to steam. If the RBCCW Pump is allowed to start, the initiation of flow would cause water hammer and likely rupture the CAR Cooler tubes.

A is incorrect. Disabling the "A" EDG will result in the unnecessary loss of one complete facility, and is NOT procedurally directed. Plausible; If the reason for not starting the RB pump is confused with SW, the EDG cannot be run.

B is incorrect. Starting the "A" Diesel Generator will cause the associated RBCCW Pump to start. With Containment greater than 20 psig, the associated CAR Coolers may be damaged due to water hammer when RBCCW flow is restored.
Plausible; This is the correct action, if CTMT pressure is below 20 psig.

C is incorrect. Placing the "A" RB Pump in Pull-To-Lock will prevent water hammer damage to the CAR coolers when the "A" EDG is started. However, this is not a proceduralized action in EOP-2525 and, therefore, is not allowed.
Plausible; This is the action that would be taken if the EDG were not started for a substantial time (when CTMT pressure drops below 20 psig).

References

EOP 2525, Rev. 025, Standard Post Trip Actions, Page 5 0f 26, Contingency Action 2.c.1

Comments and Question Modification History

12/3/10, Chip Griffin, Question #1 and #54 are similar. Replace Question #1.

02/02/11; Fixed typo in stem, "**25B** and 25B are deenergized" becomes "**25A** and 25B are deenergized". - 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

Question #: 54 | Question ID: 4000022 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 1 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

Question #: 55 | Question ID: 1100026 | ☑ RO ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

While operating at 100% power, an automatic plant trip occurs. While carrying out EOP-2525, Standard Post Trip Actions, the operators observe the following plant conditions:

- All CEAs are inserted.
- A loss of Off-Site power occurred immediately after the SIAS.
- Buses 24C and D are being supplied by their respective Diesel Generators.
- Pressurizer level is off scale low.
- Pressurizer pressure is 1410 psia, and slowly lowering.
- SIAS, CIAS, EBFAS, and MSI has properly actuated.
- Tavg is 531 °F and stable.
- Steam Generator (S/G) pressures are 890 psia and steady.
- S/G levels are both ~30% and rising.
- SJAE and Blowdown Rad. Monitors are steady.
- CTMT pressure is 4.8 psig and rising.
- CTMT Sump level indicates 100%.
- CTMT Personnel Access Rad. Monitor is rising.

Which of the following will provide circulation of the Containment Atmosphere for this event when EOP 2525 is complete?

☐ **A** Both Auxiliary Recirculation Fans must be started in slow speed.
Both Post Incident Recirculation Fans must be started.
All CAR Fans are running in slow speed.

☐ **B** Both Auxiliary Recirculation Fans are unavailable.
Both Post Incident Recirculation Fans must be started.
All CAR Fans must be manually shifted to slow speed.

☐ **C** Both Auxiliary Recirculation Fans are unavailable.
Both Post Incident Recirculation Fans will be unavailable.
All CAR Fans are running in slow speed.

☑ **D** Both Auxiliary Recirculation Fans are unavailable.
Both Post Incident Recirculation Fans must be started.
All CAR Fans are running in slow speed.

Justification

D is correct. A small break LOCA with an LNP should be diagnosed. Although required to be started in EOP 2525, Standard Post Trip Actions, the Auxiliary Recirculation Fans are NOT available because they are powered from non-vital buses which are lost as a result of the loss of off-site power. EOP 2525, Standard Post Trip Actions, require the PIR Fans to be started (with the conditions provided, both of them are available.) All CAR Fans receive a SIAS signal to start in or shift to slow speed.

A is incorrect. Auxiliary Recirculation Fans are NOT available due to the LNP; therefore, they cannot be started in slow (or fast) speed.
Plausible: The examinee may not remember that the Aux Recirc Fans are non-vital powered. Additionally, EOP 2525, Standard Post Trip Actions, requires the Aux Recirc Fans to be manually started in slow speed on high Containment temperature or pressure.

B is incorrect. The CAR Fans will automatically shift to slow speed on a SIAS.
Plausible: The examinee may think that the loss of power may cause the CAR Fans to remain running in Fast speed. A failure of the associated actuation module or a loss of the associated Vital Instrument bus will result in a CAR Fan remaining in Fast speed.

C is incorrect. Both PIR Fans are available because they are vital powered.
Plausible: The examinee may think that the PIR Fans are non-vital powered, like the Aux Recirc Fans.

References

1. AOP 2502, Rev. 004-09, Loss of Non-Vital 4.16 kV Bus 24A, Attachment 5 (Aux Recirc Fan power supply)
2. EOP 2525, Rev. 024, Standard Post Trip Actions, Steps 7 and 8.

Comments and Question Modification History

12/3/10, Chip Griffin, Is EOP 2525 Complete?
Added, "when EOP 2525 is complete" to the stem.

NRC K/A System/E/A System 103 Containment System

Generic K/A Selected

Question #: **55** | **Question ID:** 1100026 | ☑ RO | ☐ SRO | ☐ **Student Handout?** | ☐ **Lower Order?**

Rev. 0 | ☑ **Selected for Exam** | **Origin:** New | ☐ **Past NRC Exam?**

NRC K/A Generic **System** 2.2 Equipment Control

Number 2.2.44 **RO** 4.2 **SRO** 4.4 **CFR Link** (CFR: 41.5 / 43.5 / 45.12)

Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Question #: 56 | Question ID: 4054172 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 1 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

A plant startup is in progress with reactor power at 16% and Group 7 CEAs at 128 steps. The RPS Linear Nuclear Instrument (NI), Channel 'D', suddenly fails high.

What effect will this have on the Control Element Drive System (CEDS)?

☑ **A** A CEA Motion Inhibit will be generated for all regulating CEAs because of the Group 7 position when Channel 'D' failed.

☐ **B** A CEA Withdraw Prohibit will be generated for Group 7 because of the indicated high power level on Channel 'D' NI.

☐ **C** A CEA Group 7 PDI Limit annunciator will be generated by the Plant Process Computer, but CEA motion will NOT be impacted.

☐ **D** CEAs can NOT be moved in 'Manual Sequential' due to a loss of Sequential Permissive from the PPC on the abnormal core tilt.

Justification

A is correct. The Power Dependent Insertion Limit (PDIL) setpoint is based on the highest NI or Delta-T power from the four RPS channels. When channel "D" NI failed high, it caused the PDIL setpoint to "fail" to the 100% value of ~ 135 steps. This resulted in a CMI, which stops ALL rod motion.

B is incorrect. A CWP requires a 2/4 High Power or Thermal Margin/Low Pressure (TM/LP) pretrips.
Plausible: When Channel "D" fails high, high power and TM/LP pretrips are generated for that channel. The examinee may believe that a pretrip on only one channel will generate a CWP.

C is incorrect. CEA motion will be stopped by a CMI caused by the PDIL on Group 7 caused by one channel failing high.
Plausible: Even though a CEA Group 7 PDI Limit annunciator will be generated by the Plant Process Computer, the examinee may not recognize that a CMI is generated due to the Group 7 position (normal for this condition) and one channel failing high; there he/she may believe that CEA motion is unaffected.

D is incorrect. The loss of Sequential Permissive, generated by the PPC, will result in the inability to move CEAs in Manual Sequential; however, an abnormal core tilt generated by a failure of a linear power range channel will NOT cause a loss of Sequential Permissive.
Plausible: It would be logical for an abnormal core tilt to stop CEA motion; however, there is NO interlock between core tilt and CEA motion.

References

1. NIS-01-C, Rev. 5, Change 2, Nuclear Instrumentation, Page 44 of 72, second paragraph.
CED-01-C, Rev. 4, Control element Drive System, Page 32 of 68, Paragraph h.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 001 Control Rod Drive System

Number K4.07 **RO** 3.7 **SRO** 3.8 **CFR Link** (CFR: 41.7)

Knowledge of CRDS design feature(s) and/or interlock(s) which provide for the following: Rod stops

Question #: 57 | Question ID: 1100027 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant was at 100% power with the Reactor Regulating System (RRS) in a normal alignment when the Th input from loop 2 suddenly failed from 593°F to 533°F.

Assuming the initial value of Tavg was 569°F, what is the post-event indicated value of Tavg and what is the impact of this failure without operator actions?

☐ **A** Indicated Tavg on C-04 is 554°F
On a subsequent plant trip, the Condenser Steam Dump valves will close at a higher RCS temperature, forcing the ADVs to remain open longer.

☑ **B** Indicated Tavg on C-04 is 554°F
Letdown Flow will immediately go to the maximum allowed by the Letdown Limiter, due to the lower indicated Tavg.

☐ **C** Indicated Tavg on C-04 is 569°F
The Foxboro IA will substitute a Loop 2 Th value of 593°F, causing all of the condenser steam dumps to remain open longer on a subsequent plant trip.

☐ **D** Indicated Tavg on C-04 is 569°F
The Foxboro IA will automatically deselect the failed loop 2 Th, resulting only in a Foxboro DCS System Trouble alarm.

Justification

B is correct. Indicated Tavg is calculated by: Loop 1 Th + Loop 2 Th + Loop1 Tc + Loop 2 Tc / 4. With loop 2 Th at 533, the calculated Tavg is: 593°F + 533°F + 545°F +545°F / 4 = 554°F. With indicated Tavg lowering to 554°F, program PZR level lowers to 57%. The PLCS will increase letdown to the upper limit to restore actual level to the new program level.

A is incorrect. The calculated Tavg is correct; however, on a subsequent plant trip, Tavg will be very close to the normal post-trip value (both Th instruments should read close to 533°F). The Condenser Steam Dump valves should operate normally.
Plausible: The examinee may believe that the drop in Tavg was not enough to lower PZR setpoint (starts lowering at 80% power Tavg), however the Condenser Steam Dump valves will close sooner after a plant trip due to a lower than normal indicated Tavg.

C is incorrect. The indicated (calculated) Tavg is incorrect. The Loop 2 Th value will NOT have a substitute fixed value because the Foxboro system only "deselects" the failed value if the failure is of sufficient magnitude.
Plausible: The Foxboro IA is programmed to automatically substitute the other loop Thot on an instrument failure. The examinee may believe that indicated Tavg will remain at a higher value than actual Tavg on a plant trip due to the "substituted" value. This would cause the Steam Dump valves to remain open.

D is incorrect. The indicated (calculated) Tavg is incorrect. The failed Loop 2 Th will NOT be deselected because it did not fail low enough (</= 513°F, not 533°F).
Plausible: Although the Loop2 Th instrument may be manually deselected (procedurally directed action), the examinee may believe that the failed instrument is automatically deselected.

References

RRS-01-C, R4, Pg. 18, Abnormal Operation , Thot Failures

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 016 Non-Nuclear Instrumentation System (NNIS)

Number A3.02 **RO** 2.9* **SRO** 2.9* **CFR Link** (CFR: 41.7 / 45.5)

Ability to monitor automatic operation of the NNIS, including: Relationship between meter readings and actual parameter value

Question #: 58 | Question ID: 1100028 | ☑ RO ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

A Steam Generator Tube Rupture has occurred and the crew has entered EOP-2534. The crew has begun a plant cool down using Natural Circulation and the RO is evaluating RCS subcooling using the Plant Process Computer (PPC).
Presently, both channels of ICC indicate 35°F subcooled on the PPC.

Then, a CET on Channel "A" suddenly fails to 900°F.

Which of the following describes the expected response of the displayed values for subcooling?

☐ **A** The PPC will automatically deselect the failed CET and calculate "CET max" and "CET high" subcooling for Channel "A" based on the next highest two CETs.

☐ **B** The PPC will automatically deselect the failed CET and calculate both "CET max" and "CET high" subcooling for Channel "A" based on the second highest CET.

☑ **C** The PPC will continue to use the failed CET to calculate "CET max" subcooling for Channel "A" and update the value accordingly as RCS pressure changes.

☐ **D** The PPC will continue to use the failed CET to calculate "CET max" subcooling for Channel "A" but will not update the value as RCS pressure changes.

Justification

C - CORRECT; The PPC uses the highest in-service CET to calculate a "maximum subcooling value" based on all CETs in a given channel. If a CET fails to a higher value than all the others in that channel, then the CET Max subcooling value will be calculated using that CET until it is taken out of service.

A - WRONG; The PPC will NOT detect an abnormally high CET and remove it from service unless outside the range of 32°F to 2300°F.
Plausible: The examinee may believe this because this is how the Foxboro IA system functions when temperature detectors fail.

B - WRONG; "CET Max" is NOT calculated using the second highest CET value for Channel "A" unless the highest CET value is manually taken out of scan.
Plausible: "CET High" is the subcooling value for the second highest CET in a channel. The examinee may believe that an abnormally high CET reading on Channel "A" will be automatically removed from scan by the PPC and replaced with the second highest CET reading.

D - WRONG; The PPC does NOT automatically "freeze" a calculated subcooling value based on a failed CET.
Plausible: The examinee may believe that an abnormally high CET reading on Channel "A" will be automatically locked in by the PPC when a failure is recognized, because the Foxboro IA displayed values can sometimes respond in this fashion when inputs fail due to a loss of instrument power.

References

ICC-00-C, R1C1, Pg 16, CET System Design and Operating Characteristics.

Comments and Question Modification History

02/02/11; Changed CET failure value from "**1000**°F" to "**900**°F" and changed "**ignore**" to "**deselect**" in choices 'A' and 'B'. - rlc.

NRC K/A System/E/A **System** 017 In-Core Temperature Monitor System (ITM)

Number K6.01 **RO** 2.7 **SRO** 3.0 **CFR Link** (CFR: 41.7 / 45.7)

Knowledge of the effect of a loss or malfunction of the following ITM system components: Sensors and detectors

Question #: 59 | Question ID: 1180021 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The plant is stable in Mode 1 with all systems and components functioning as designed.

Containment pressure is 18" of water and decreasing due to venting using the H2 purge valves and EBFS to the site stack.

If an RCS leak were to occur in containment, which of the following conditions would automatically terminate the release?

☐ **A** Radiation alarms on any of the gaseous or particulate containment atmospheric monitors.

☐ **B** Containment atmosphere radiation triggering an alarm on the Kaman stack rad. monitor.

☐ **C** Containment pressure exceeding 1.0 psig on the C-01 narrow range pressure indication.

☑ **D** Containment pressure on 2 or more wide range indications on C-01 exceeding 4.5 psig.

Justification

D - CORRECT; This is the CIAS setpoint, which when actuated on rising CTMT pressure would automatically close the purge valves.

A - WRONG; A high CTMT radiation signal is required to close the dampers automatically, not an alarm on the CTMT atmospheric monitors.
Plausible; It is logical that a high rad. alarm on a CTMT atmospheric monitors would isolate the CTMT purge valves, especially when it does isolate the CTMT ventilation system dampers if they were open.

B - WRONG; An alarm on the Kaman rad monitor does not cause the purge valves to isolate. It just purges the normal stack rad monitor.
Plausible; The Kaman would alarm on a high radiation release to the environment and a high radiation condition in CTMT is what does close the purge valves automatically.

C - Wrong; CTMT pressure must reach the SIAS/CIAS setpoint to trigger an isolation.
Plausible; This is the Tech. Spec. limit for CTMT pressure, where operator action to depressurize is required.

References

CSS-01-C, R10C1, Pg. 31 of 82, 17. "EBFS Dampers", last paragraph, CTMT Purge Isolation Valves receive a CIAS closure.

Comments and Question Modification History

12/3/10, Chip Griffin, 18" water *gravity* in the stem. Seems a bit odd to use the word gravity. Not normally used.
Removed the word 'gravity'.

02/02/11; Per validation, change 'D' pressure value from "**3.75** psig" to "**4.5** psig". - rlc.

NRC K/A System/E/A **System** 028 Hydrogen Recombiner and Purge Control System (HRPS)

Number A4.02 **RO** 3.7* **SRO** 3.9 **CFR Link** (CFR: 41.7 / 45.5 to 45.8)

Ability to manually operate and/or monitor in the control room: Location and interpretation of containment pressure indications

Question #: N/A | Question ID: 8000021 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 1 | ☐ Selected for Exam | Origin: Parent | ☑ Past NRC Exam?

PARENT

The plant tripped due to a Large Break LOCA several hours ago and the following conditions now exist:

- EOP-2532 in progress
- All plant systems and components functioning as designed for the LB-LOCA
- Containment pressure = 2.5 psig and lowering slowly
- Containment Spray has been secured per procedure, based on improving containment conditions
- Containment Hydrogen purge in progress
- All other plant systems and components functioning as designed

Then, the break in the RCS gets larger, causing Containment pressure to start rising.

Which of the following describes how the Containment Hydrogen Purge duct work will be protected from an over-pressure condition?

☑ **A** Hydrogen Purge Dampers must be manually closed from their C-01 control switches.

☐ **B** Hydrogen Purge Dampers will automatically close on rising radiation levels in CTMT.

☐ **C** Hydrogen Purge Dampers will close by manual or automatic re-actuation of a CSAS.

☐ **D** Hydrogen Purge Dampers will close by manual or automatic re-actuation of a CIAS.

Justification

A - Correct; The isolation valves must be overridden open and, therefore, must be closed by re-operation of their control switches.
B - Wrong; This implies that the a subsequent high CTMT radiation signal will still close the dampers, even though a CIAS signal was overridden to open them. High CTMT radiation also closes these dampers, and this condition would be present. However, this is NOT correct because high radiation must already exist in CTMT for the excessive generation of hydrogen to take place (severe fuel damage). Therefore, the Hi Rad signal must have already been overridden to initially open the dampers.
C - Wrong; Although CSAS is "reset" when spray is secured, and it will automatically re-actuate on rising CTMT pressure, this signal will NOT automatically close the purge isolation valves.
D - Wrong; If CIAS where to re-actuate on rising CTMT pressure it would automatically close the purge isolation valves. However, the given conditions do NOT state that the CIAS signal has been reset and, unlike the direction given to secure and reset CSAS, EOP-2532 does NOT direct CIAS be reset on improving CTMT conditions. Although a SIAS/CIAS signal is required for ESAS to generate a CSAS, the SIAS/CIAS signal does NOT have to be reset in order to reset the CSAS signal. Unlike CIAS, which requires SIAS be reset on ESAS in order to reset a CIAS signal.

References

LP CSS-01-C, CTMT Purge Isolation Valves w/ CIAS.

Comments and Question Modification History

Modified question per NRC comments; 02/19/09
- Replaced specific ESAS signals that actuated in stem with "All plant systems and components functioning as designed .."
- Changed accident from LOCA to LB-LOCA to improve plausibility of hydrogen generation.
- Added clarification to choice "D" justification to improve understanding of plausibility.

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)

Question #: 60 | Question ID: 78242 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 1 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

New Spent Fuel Pool Cooling Heat Exchangers are being installed (one at a time) during MODE 1, EOL operation. The "A" heat exchanger was dropped on the piping for the "B" heat exchanger causing a loss of all RBCCW to the SFP cooling system. SFP temperature is 120°F and rising. Maintenance predicts it will take 4 to 6 days to restore at least one SFP Heat exchanger to service.

Which of the following methods of cooling the Spent Fuel Pool will be utilized?

☐ **A** Supply Primary Makeup Water to the Spent Fuel Pool to make up for losses due to evaporation.

☐ **B** Cross-tie Shutdown Cooling with Spent Fuel Pool Cooling and start a LPSI Pump

☑ **C** Fill the Spent Fuel Pool to the high level then drain to the low level using the RWST.

☐ **D** Cross-tie Shutdown Cooling with Spent Fuel Pool Cooling and start a Containment Spray Pump.

Justification

C - CORRECT; AOP 2582 provide 4 methods to maintain SFP cooling if SFP cooling is lost or not available. The first method listed is to use SDC; however, this is supplemental cooling and requires SDC to be in service. Obviously, SDC is NOT in service in MODE 1; therefore, the next listed method is to fill the SFP from the RWST to the high level alarm, then drain it back to the RWST to the low level alarm. This method of cooling utilizes the RWST as the heat sink and may be used until the RWST reaches its upper temperature limit. The other two methods employ the use of PMW or Aux Feed to fill the SFP then drain to the Clean Waste Tank. This method will result in creating rad waste and cannot be used for an extended period of time due to waste management issues.

A is incorrect. Even though it may provide some cooling for the SFP, PMW to make up for losses is NOT an approved method. Plausible: While this will help keep the SFP cooled, it is NOT procedurally approved. The examinee may NOT remember the all approved SFP cooling methods.

B is incorrect; Cross-tying SDC with SFP cooling is the preferred method of cooling the SFP when SFP cooling is NOT available; however, SDC must be in service.
Plausible: The examinee may remember that AOP 2582, Loss of SFP Cooling provides guidance for cross-tying SDC with SFP Cooling; however he/she may NOT remember that SDC must be in service. The examinee may think that it is ok to use a LPSI Pump with a SDC Heat Exchanger for SFP cooling any time.

D is incorrect. Cross-tying SDC with SFP cooling is the preferred method of cooling the SFP when SFP cooling is NOT available; however, SDC must be in service. (A Containment Spray Pump may be substituted for a LPSI Pump.)
Plausible The examinee may remember that AOP 2582, Loss of SFP Cooling provides guidance for cross-tying SDC with SFP Cooling; however he/she may NOT remember that SDC must be in service. AOP 2582 allows a Containment Spray Pump to used in place of a LPSI Pump.

References

AOP-2582, R2C3, Pg. 6 of 22, St. 4.1.6.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 033 Spent Fuel Pool Cooling System (SFPCS)

Number K1.05 **RO** 2.7* **SRO** 2.8* **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 Spent Fuel Pool Cooling System and the following systems: RWST

Question #: 61 | Question ID: 1100029 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

Given the following conditions:

- The plant is at 95% power, having just started up following a refueling outage.
- All systems are in normal lineup to support 100% power operation.
- CONVEX orders an Emergency Generation Reduction to 580 MWe within the next 15 minutes.
- The crew initiates AOP 2557, "Emergency Generation Reduction"

While performing the load reduction, generator output is lowered a little too quickly causing RCS temperature and SG pressure to suddenly rise. Both SG levels initially spiked down when SG pressure rose, but are now back to normal.

Which of the following reactivity and parameter changes would result from the above conditions?

☐ **A** Negative reactivity addition due to rising main feedwater flow.

☑ **B** Negative reactivity addition due to rising RCS temperature.

☐ **C** Positive reactivity addition due to rising main feedwater flow.

☐ **D** Positive reactivity addition due to rising RCS temperature.

Justification

B - CORRECT; Even though the core is at BOL conditions, at this power level MTC would still be negative. Therefore, rising RCS temperature will add negative reactivity.

A - WRONG; The drop in SG level was due to shrink when steam demand was suddenly reduced (causing the rise in RCS temperature and SG pressure). The drop in steam flow would counter the momentary drop in SG level seen by the SG Water Level Control System (SGWLC). Therefore, feedwater flow would not rise enough to effect RCS temperature or reactivity.
Plausible; Examinee may believe the SG levels returning to normal was due to rising main feed flow, which would cause a drop in RCS temperature. At low power BOL conditions (positive MTC), a drop in RCS temperature would result in a negative reactivity addition. The emergency Generation Reduction does NOT require reactor power to be lowered, only Turbine load.

C - WRONG; Main feed flow will not rise enough to effect RCS temperature because the MFW system will see the drop in steam flow that caused the drop in SG level (shrink).
Plausible; Examinee may believe the SGWLC system would respond to the drop in SG level and raise feed flow to restore it (as indicated). Rising feed flow would cause a subsequent drop in RCS temperature and, with a positive MTC, cause a positive reactivity addition.

D - WRONG; Even though the reactor is at BOL conditions, at this power level, MTC would be negative.
Plausible; Examinee may believe that because the core is at BOL conditions it would have a positive MTC. (MTC is positive at low power conditions BOL.)

References

1. RE Curve and Data Book, Moderator Temperature Coefficient Versus Boron Concentration, RE-G-03
2. Reactivity Imbalances LP, RIB-01-C
3. Admin Controls: Reactivity Management, ADM-01-C

Comments and Question Modification History

Reworded the question statement in the stem to clarify what was being asked, per comment from Sandy Doboe. 01/06/11

NRC K/A System/E/A **System** 035 Steam Generator System (S/GS)

Number K5.01 **RO** 3.4 **SRO** 3.9 **CFR Link** (CFR: 41.5 / 45.7)

Knowledge of operational implications of the following concepts as the apply to the S/GS: Effect of secondary parameters, pressure, and temperature on reactivity

Question #: 62 | Question ID: 1100030 | ☑ RO | ☐ SRO | ☑ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The following stable plant conditions exist:
- The plant is at 80% power
- Tc is 544.5°F (2.5°F above program temperature)
- Present Burnup is 8500 MWD/MTU
- Present RCS Boron concentration is 700 ppm
- Inverse Boron Worth is 112 ppm/%Δρ

The BOP raises Turbine load to restore Tc to program temperature.

Considering ONLY the affects of Moderator Temperature, which of the following describes the value of the Reactivity change caused by the change in RCS temperature and the required change to the RCS Boron concentration to maintain power at 80%?

☐ **A** -0.040%Δρ
Add 54 gallons of Boric Acid

☐ **B** -0.040%Δρ
Add 390 gallons of PMW

☑ **C** +0.035%Δρ
Add 47 gallons of Boric Acid

☐ **D** +0.035%Δρ
Add 341 gallons of PMW

Justification

C is correct. Using the Reactivity Thumb Rules (provided), Moderator Temperature Coefficient is -0.014%Δρ/°F. (-0.014%Δρ/°F x -2.5°F = +0.035%Δρ) Because positive reactivity is added when Tc is lowered, Boric Acid must be added to compensate and maintain power at 80%. The thumb rule states that 12 gallons of Boric Acid must be added for every ppm rise in RCS Born concentration. It was also given that Inverse Boron Worth is 112 ppm/%Δρ. (0.035%Δρ x 112 ppm/%Δρ x 12 gal/ppm increase in RCS Boron = 47 gallons of Boric Acid) Another method using the Reactivity Thumb Rules: (+0.035%Δρ / +0.016%Δρ/% pwr change = +2.1875% pwr change. +2.1875% pwr change x 1.8 ppm Boron change/% pwr change = 3.9375 ppm Boron increase. 3.9375 ppm x 12 gal/ ppm = 47.25 gallons of Boric Acid)

A is incorrect. The Power Defect, as given on the Reactivity Thumb Rules, is 0.016 %Δρ/% power change. (0.016 x -2.5 = -0.040%Δρ) If this answer were used, then 54 gallons of Boric Acid would need to be added. (-0.040%Δρ x 112 ppm/%Δρ x 12 gal/ppm increase in RCS Boron = -54 gallons of Boric Acid)
Plausible: If the examinee confuses the reactivity added from the power defect instead of the reactivity added by ONLY the change in Moderator Temperature and neglects or confuses the (+, -) sign, then he/she may use the Power Defect from the Reactivity Thumb Rules. The examinee may realize that a lower moderator temperature requires Boron to be added.

B is incorrect. The Power Defect, as given on the Reactivity Thumb Rules, is 0.016 %Δρ/% power change. (0.016 x -2.5 = -0.040%Δρ) If this answer were used, then it would indicate that negative reactivity was inserted a PMW must be added to lower RCS Boron concentration. The Reactivity Thumb Rules states that 87 gallons of PMW must be added for every ppm reduction in RCS Boron. (0.040%Δρ x 112 ppm/%Δρ x 87 gal/ppm decrease in RCS Boron = 390 gallons of PMW).
Plausible: If the examinee confuses the reactivity added from the power defect instead of the reactivity added by ONLY the change in Moderator Temperature, then he/she may use the Power Defect from the Reactivity Thumb Rules. The calculation produces a negative reactivity from the temperature change which requires the addition positive reactivity from PMW.

D is incorrect. Although +0.035%Δρ is the appropriate value of reactivity added by reducing temperature, adding PMW would result in a further rise in power.
Plausible: During any power ascension, when Turbine load is raised, PMW is also added (or CEAs are withdrawn) to continue raising power. If the examinee confuses a normal evolution (raising load) with this evolution, then he/she may believe that adding PMW is appropriate. Additionally, the examinee may be confused by the + sign which may indicate that positive reactivity must be added.

References **Provided**

Provide OP 2208, Attachment 5, Reactivity Thumb Rules for 8500 MWD/MTU.

Comments and Question Modification History

01/20/11; Annotated question as requiring Handout during exam, per References field. - rlc.

NRC K/A System/E/A **System** 045 Main Turbine Generator (MT/G) System

Question #: **62** Question ID: 1100030 ☑ RO ☐ SRO ☑ Student Handout? ☐ Lower Order?

Rev. 0 ☑ Selected for Exam Origin: New ☐ Past NRC Exam?

Number K5.17 **RO** 2.5* **SRO** 2.7* **CFR Link** (CFR: 41.5 / 45.7)

Knowledge of the operational implications of the following concepts as the apply to the MT/B System: Relationship between moderator temperature coefficient and boron concentration in RCS as T/G load increases

Question #: 63 | Question ID: 2000033 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 1 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

The plant is tripped from 100% power due to a Steam Generator Tube Rupture and all equipment responded as designed on the trip.
The actions of EOP-2525 have been carried out and the crew has just transitioned to EOP-2534, SGTR.
A few minutes after transitioning, the RO reports that SIAS, CIAS and EBFAS have actuated on pressurizer pressure and all equipment has responded as expected.
The STA subsequently notices that condenser vacuum is degrading.

Which of the following describe actions that must be taken to maintain condenser vacuum?

☐ **A** Override and re-open 2-EB-55 & 2-EB-56, condenser air removal isolation dampers to the site stack.

☐ **B** Swap condenser air removal to the high flow fan, MF-55A, and open EB-171, MF-55A makeup damper.

☐ **C** Open 2-MS-182, gland sealing steam bypass valve, and restore gland sealing steam pressure to normal.

☑ **D** Open 2-EB-57, condenser air removal to Unit 2 stack isolation damper, and start one main exhaust fan.

Justification

D; CORRECT; opening EB-57 provides Condenser Air Removal (CAR) fan flow path, which was automatically isolated by the CIAS, backing up non-condensibles in the main condenser.

A - WRONG; EB-55 & 56 automatically close on EBFAS, reopening would parallel CAR fan with EBFS for the discharge path, not procedurally allowed and wouldn't work.
Plausible; Discharging radioactive steam (SGTR) out the station stack is considered an "air release" and is preferred to discharging out the Unit 2 stack, which is considered a "ground release".

B - WRONG; no discharge flow path is available for the either C.A.R. fan unless EB-57 is opened.
Plausible; Higher capacity fan may overcome EBFAS fan back pressure.

C - WRONG; gland seal steam never was interrupted by the given ESAS signals. This would automatically happen on a MSI signal.
Plausible; The normal gland seal regulator is know to stick closed on a trip as it is not open above ~20% power (glands self-seal then).
This would be the expected action if the stem did not state that all equipment functioned as designed.

References

EOP-2534, R25; Pg. 11, St. 7, Align Cndsr Air Removal to U-2 Stack.

Comments and Question Modification History

Modified "and verify a main exhaust fan is running normally." to "and ensure at least one main exhaust fan is running." per comments from Sandy Doboe. 01/06/11

NRC K/A System/E/A **System** 055 Condenser Air Removal System (CARS)

Number A3.03 **RO** 2.5 **SRO** 2.7* **CFR Link** (CFR: 41.7 / 45.5)

Ability to monitor automatic operation of the CARS, including: Automatic diversion of CARS exhaust

Question #: 64 | Question ID: 1110110 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

A plant startup is in progress with power presently at 99% and all equipment operating normally.
One of the three running condensate pumps then trips on overload.

Which of the following describes the effect of the loss of a condensate pump on the secondary system and the appropriate action to be taken?

☐ **A** The loss of a condensate pump will drop Main Feed Pump suction pressure and affect the supply to the SGs. Take manual control of both Main Feed Pumps and maintain their speed constant.

☑ **B** The loss of a condensate pump will drop Main Feed Pump suction pressure and affect the supply to the SGs. Throttle open the CPF Bypass Valve to restore Main Feed Pump suction pressure.

☐ **C** The lower condensate flow will cause cavitation in the Heater Drains Pumps due to the higher heater drains flow. Open HD-106, feed water heater drains subcooling valve and monitor heater drain tank level.

☐ **D** The lower condensate flow will cause cavitation in the Heater Drains Pumps due to the higher heater drains flow. Throttle open the CPF Bypass Valve to restore Main Feed Pump suction pressure.

Justification

B - CORRECT; The plant was originally designed to operate on two condensate pumps, but only without the flow restriction (pressure loss) of the CPF demineralizers. Therefore, at this power level, two condensate pumps cannot supply adequate suction pressure to the feed pumps without bypassing CPF.

A - WRONG; Although automatic SGFP control will speed up the pumps in an attempt to maintain FRV delta-P constant, this action will backfire if it is the only one taken and results in a loss of SG level control.
Plausible; A loss of a Condensate Pump results in lower feed pump suction pressure and a reduction in feed flow. The examinee may believe that maintaining feed pumps speeds constant in manual (vs an automatic speed increase) will allow the automatic operation of the Main Feed Reg Valves to maintain S/G level and maintain adequate Feed Pump suction pressure.

C - WRONG; Opening HD-106 would divert more condensate pump discharge flow from the SGFPs and make conditions worse.
Plausible; This is the correct action if it were a Heater Drain Pump that tripped at this power level.

D - WRONG; Flow will not rise sufficiently high to cause cavitation with both Heater Drain Pumps operating.
Plausible; Heater drains flow will rise substantially with the loss of a condensate pump at this power level.

References

1. CON-00-C, R8, Pg. 13
2. ARP-2593E-017, R0C0, CPF System Delta-P High, Step 5

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 056 Condensate System

Number A2.04 **RO** 2.6 **SRO** 2.8* **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 Condensate System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations Loss of condensate pumps

Question #: N/A | Question ID: 1000110 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 2 | ☐ Selected for Exam | Origin: Parent | ☑ Past NRC Exam?

PARENT

A reactor trip occurred from 100% power. On the plant trip, the BOP notes that both 6900 volt AC buses are deenergized due to a failure to transfer to the RSST. All other electrical buses are energized from their normal source.

Which of the following describes the effect of the loss of power on the secondary system and the appropriate action to be taken?

☑ **A** Condenser vacuum will be lost due to the loss of cooling to the steam jet air ejector. Per EOP 2525, Standard Post Trip Actions, the BOP must close the MSIVs and open 2-AR-17, Condenser Vacuum Breaker.

☐ **B** The loss of cooling water to the Gland Exhaust Condenser will cause a loss of condenser vacuum. Per Appendix 4, the BOP must manually control steam seal pressure using MS-182A, Gland Seal supply bypass.

☐ **C** Excessive water hammer in the feedwater heaters will begin due to the loss of condensate flow. Once Standard Post Trip Actions are completed, the BOP must close all extraction steam supply valves on C06/7.

☐ **D** The loss of Main Feed supply to the SGs will cause hotwell reject to overflow the condensate surge tank. Once Standard Post Trip Actions are completed, the BOP must instruct Ionics to secure makeup flow.

Justification

A - Correct; A loss of 6900 Volt AC buses results in a loss of condensate pumps. EOP 2525, step 11 states that if offsite power is lost or no condensate pumps are operating, then the condenser is NOT available; close both MSIVs and open 2-AR-17, condenser vacuum breaker.
B - Wrong; the turbine seals will still be maintained by gland seal steam until the MSIVs are closed. The gland seal supply valves will automatically throttle to maintain the appropriate steam seal pressure.
C - Wrong; On a loss of condensate pumps, the subsequent actions of EOP 2525 require the BOP to close the MSIVs, which will isolate steam flow to the feedwater heaters as much as is practical. Although this will NOT eliminate all hydraulic stress (lots of noise) seen by the feedwater heaters due to the sudden loss of condensate and feed flow, this is all that can be done at this time.
D - Wrong; Because condensate pump discharge pressure provides flow to the CST through the "reject" valve, the loss of 6900 volt AC buses causes a loss of condensate pumps; therefore the CST will NOT fill up.

References

LP CAR-00-C, R-3, C-5, Condenser Air Removal, (three linked locations)
EOP-2525, R-23, St. 10 (Inst & Cont)

Comments and Question Modification History

Per NRC comments, the following changes were made: 02/18/09
1) Corrected justifications for choices "A" & "B".
2) Removed "During the performance of EO-2525, Standard Post Trip Actions" from the stem.
3) Reworded choice "B" to reference actions taken per App. 4, controlling of gland seal manually.
4) Replaced choice "C" required action (limit AFW flow based on high hotwell level).
5) Replaced choice "D" required action (close the reject valve to the surge tank).

NRC K/A System/E/A System E02 Reactor Trip Recovery

Number EA1.1 **RO** 3.7 **SRO** 3.7 **CFR Link** (CFR: 41.7 / 45.5 / 45.6)

Ability to operate and/or monitor components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features as they apply to the Reactor Trip Recovery.

Question #: 65 | Question ID: 1100031 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

OP 2325D, Backwashing Operations, is being performed with the following conditions:
- Thermal Backwashing is scheduled in "A" Circ Bay first.
- All steps of Section 4.1, Initial Actions for Thermal Backwashing and Backwashing Operations, are complete.
- "B" Service Water Pump is in operation.
- The tide is nearly High and outgoing.
- Injection Temperature is 60°F.

Which of the following statements describes a potential plant impact from Thermal Backwashing Operations?

☑ **A** An operator must be stationed at the Vital Switchgear inlet temperature gage to determine if the Ultimate Heat Sink temperature limit is exceeded.

☐ **B** An operator must be stationed in the Intake structure to monitor Lube Water flow to ensure Circulating Water Pump bearing flows remain within limits.

☐ **C** Sodium Hypochlorite flow to the "B" Service Water Pump must be raised to kill the mussels in the "A" Circulating Water Bay during Thermal Backwashing.

☐ **D** Radioactive liquid waste discharges are NOT allowed to be performed during mussel cooking operations to ensure compliance with the NPDES permit.

Justification

A is correct. The water in the bay being mussel cooked is heated and flows out the front of at bay and is drawn into the adjacent bays. This results in the adjacent bays, which have running Service Water Pumps, heating up and reducing the effectiveness of cooling. During periods of elevated Intake temperatures (>70°F), an operator is required to be stationed at the Vital Switchgear inlet temperature gage to monitor Service Water inlet temperature. Tech Spec LCO 3.7.11, Ultimate Heat Sink, must be entered if the Service Water inlet temperature to the Vital Switchgear Coolers exceeds 74.5°F.

B is incorrect. An operator is NOT required to specifically monitor the Service Water Lube Water flow to the Circulating pump bearings during Mussel cooking.
Plausible: If Service Water inlet temperature rises then the heat exchangers with Temperature Control Valves will require more Service Water flow which causes a reduction in Service Water Header pressure. A lower pressure will result in lower low to components served by Service Water. The Circulating Water Pump Lube Water pressure is controlled by a pressure control valve which maintains pressure at approximately 40 psig; therefore, flow will not change.

C is incorrect. Sodium Hypochlorite to the Service Water Pumps is secured during mussel cooking to ensure the NPDES permit is not violated by discharging Sodium Hypochlorite from an unauthorized discharge point.
Plausible: Sodium Hypochlorite is injected into the bay that has just completed mussel cooking. It would be logical to assume sodium Hypochlorite is injected during mussel cooking.

D is incorrect. Radioactive waste discharges are allowed during mussel cooking as long as the permit contains the appropriate Circulating Water configuration.
Plausible: Radioactive discharges are secured only when a <u>change</u> occurs to the Circulating Water System that is NOT accounted for in the permit. The examinee may assume a radioactive discharge is NOT allowed to continue when mussel cooking operation begins.

References

OP 2325D, R6C1, Precaution 3.2 and Step 4.2.10a, third bullet.

Comments and Question Modification History

12/3/10, Chip Griffin, Replace the term 'Mussel Cook' with 'Thermal Backwash'
Done. Replaced mussel cooking in distractor C with Thermal Backwash.

02/01/11; Per validation, changed Choice 'A' from **"ensure"** Ultimate Heat Sink temperature limit is "**NOT**" exceeded, to **"determine if"** the heat sink is exceeded, as monitoring a parameter does not in itself prevent anything. Also, changed Injection Temperature from "**71**° F" to "**60**°F" to improve realism. - rlc.

NRC K/A System/E/A **System** 075 Circulating Water System

Number K1.08 **RO** 3.2* **SRO** 3.2* **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 circulating water system and the following systems: Emergency/essential SWS

Question #: **66** | **Question ID: 1150003** | ☑ RO | ☐ SRO | ☑ Student Handout? | ☐ Lower Order?

Rev. 0 | ☑ Selected for Exam | **Origin: Mod** | ☐ Past NRC Exam?

The reactor automatically tripped from full power. The US has just entered EOP-2525, "Standard Post Trip Actions". NO operator actions have been taken.

Using the attached copy of the SPDS display, identify the major event that has occurred.

☐ **A** A stuck open Main Steam Safety Valve

☐ **B** A Small Break LOCA on the Head seal

☐ **C** A partially stuck open Pressurizer Safety

☑ **D** A small Steam Line Break inside Containment

Justification

D is correct. The lower S/G pressures and RCS temperatures while maintaining RCS subcooling are indicative of an Excess Steam Demand event. The rising Containment pressure is indicative of the ESD being inside Containment.

A is incorrect. Containment pressure is elevated; therefore, the event is an energy release inside Containment. A stuck open Main Steam Safety is an ESD outside CTMT.
Plausible: The examinee will see the classic symptoms of a stuck open safety valve but may miss the elevated Containment temperature and pressure.

B is incorrect. RCS subcooling is being maintained; therefore, the event is NOT a Small Break LOCA.
Plausible: The examinee may believe that the lower RCS temperature and pressure and lower S/G pressures are caused by Safety Injection flow due to a LOCA. Additionally, rising Containment temperature and pressure could also be attributed to a LOCA.

C is incorrect. Pressurizer level would likely rise if a Pressurizer Safety were partially open
Plausible: The examinee may believe that the abnormally low Pressurizer pressure, low (but not empty) Pressurizer level, and rising Containment pressure and temperature, coupled with a full Containment Sump, are due to a LOCA caused by a stuck open Pressurizer Safety.

References **Provided**

Reference:
E36-01-C, Excess Steam Demand Lesson Text.

Requires copy of SPDS screen after a trip with a stuck open Main Steam Safety Valve. (Other malfunctions are added to complicate the diagnosis.)

Comments and Question Modification History

12/17/10, RJA
Changed distractor B to an intersystem LOCA in the Letdown system.

NRC K/A System/E/A **System** 2.1 Conduct of Operations

Generic K/A Selected

NRC K/A Generic **System** 2.1 Conduct of Operations

Number 2.1.19 **RO** 3.9 **SRO** 3.8 **CFR Link** (CFR: 45.12)
Ability to use plant computers to evaluate system or component status.

Question #: N/A | Question ID: 5000003 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☑ Past NRC Exam?

PARENT

The reactor automatically tripped from full power. The US has just entered EOP-2525, "Standard Post Trip Actions". NO operator actions have been taken.

Using the attached copy of the SPDS display, identify the event that has occurred.

☐ **A** Feed Line Break

☑ **B** Loss of Coolant Accident

☐ **C** Steam Generator Tube Rupture on #2 SG

☐ **D** Excess Steam Demand inside Containment

Justification

CHOICE (A) - NO
WRONG: Event is LBLOCA.
VALID DISTRACTOR: Containment pressure is elevated.

CHOICE (B) - YES
SPDS display copied off of simulator 1 minute after initiating fail open of #2 FRV followed by LBLOCA concurrent with trip from full power

CHOICE (C) - NO
WRONG: Event is LBLOCA.
VALID DISTRACTOR: SG #2 level higher than SG #1 level.

CHOICE (D) - NO
WRONG: Event is LBLOCA.
VALID DISTRACTOR: Thot, Tcold much lower than normal post-trip

References

1. PPC-00-C, "Plant Process Computer System" Lesson, Revision 1 (12/22/03) (Pg 16, 17, 30 of 30)

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 011 Large Break LOCA

Number EA1.17 **RO** 3.5* **SRO** 4.1* **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to operate and monitor the following as they apply to a Large Break LOCA: Safety parameter display system

Question #: 67 | Question ID: 1100032 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

A plant shutdown from 100% power for planned maintenance is about to commence. The BOP has been directed to commence forcing Pressurizer Sprays.

Which of the following describes the appropriate method for performing this task?

☐ **A** Using the guidance in AOP 2575, Rapid Downpower:
- Observe Pressurizer pressure controller setting and present RCS pressure.
- Lower Pressure Controller to obtain a 40-50% output.
- Place all Backup Heater Group hand switches to "Close".
- Monitor RCS pressure and adjust as required.

☑ **B** Using OP 2204, Load Changes, Attachment 9, Forcing Pressurizer Sprays:
- Record Pressurizer pressure controller setting and present RCS pressure.
- Place all Backup Heater Group hand switches to "Close".
- Lower Pressure Controller to obtain a 40-50% output.
- Monitor RCS pressure and adjust as required.

☐ **C** Using SP 2654B, Forcing Pressurizer Sprays:
- Record Pressurizer pressure controller setting and present RCS pressure.
- Lower Pressure Controller to obtain a 40-50% output.
- Place all Backup Heater Group hand switches to "Close".
- Monitor RCS pressure and adjust as required.

☐ **D** Using the guidance for performing routine tasks in OP-AA-100, Conduct of Operations:
- Observe Pressurizer pressure controller setting and present RCS pressure.
- Place all Backup Heater Group hand switches to "Close".
- Lower Pressure Controller to obtain a 40-50% output.
- Monitor RCS pressure and adjust as required.

Justification

B is correct. A planned shutdown requires the use of OP 2204, Load Changes. Attachment 9 provides the guidance for forcing Pressurizer sprays.

A is incorrect. AOP 2575, Rapid Downpower is NOT used for a planned downpower.
Plausible: The examinee may feel that the guidance in AOP 2575 is the same as the guidance in OP 2204. Even though both procedures require forcing sprays for the downpower, OP 2204 actually provides more information for forcing sprays.

C is incorrect. SP 2654B provides the same guidance as OP 2204, however, forcing spays for the downpower must be performed in accordance with the governing procedure (OP 2204)
Plausible: SP 2654B provides the same level of detail as OP 2204 for forcing sprays; therefore, the examinee may not remember that OP 2204 has a specific attachment for forcing sprays.

D is incorrect. Although forcing sprays is a relatively routine task, OP-AA-100 still requires the operator to use the appropriate procedure guidance.
Plausible: The examinee may (incorrectly) feel that forcing sprays is a routine task allowed to be performed without the procedure in hand.

References

OP-2204, R23C5, Attachment 9, Forcing PZR Sprays.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.1

Generic K/A Selected

NRC K/A Generic **System** 2.1 Conduct of Operations

Number 2.1.23 **RO** 4.3 **SRO** 4.4 **CFR Link** (CFR: 41.10 / 43.5 / 45.2 / 45.6)

Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Question #: 68 | Question ID: 1178685 | ☑ RO | ☐ SRO | ☑ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

AOP 2580, Degraded Voltage, has the crew refer to Attachment 1, "Estimated Capability Curves" and ensure operation is within limits.
The main generator output is currently 700 MWe and hydrogen pressure is 60 psig.

What is the maximum MVARs lagging that the generator can produce and stay within the limits of the curve?

☐ **A** 340 MVARs

☐ **B** 360 MVARs

☐ **C** 440 MVARs

☑ **D** 570 MVARs

Justification

D - CORRECT; Per AOP 2580, Attachment 1 (**required**), at 700 MWe and 60# hydrogen, the max MVAR loading is 580 MVARs.

A - WRONG; Generator output is read on the "X" axis and the MVAR limit is read on the "Y" axis.
Plausible; Reversing the "X" and "Y" axis equates to a 340 MVAR limit.

B - WRONG; Unit 2 is required to have a "lagging" power factor.
Plausible; A "leading" power factor equates to 360 MVAR limit.

C - WRONG; The 0.90 Power Factor is a guide, not a specific limit.
Plausible; 0.90 PF line, stated in the information above the curve, equates to 440 MVAR limit.

References **Provided**

Requires use of AOP-2580, R3C4; Att. 1 Curve

Comments and Question Modification History

02/02/11; Per validation, lowered correct answer from "**580** MVARs" to "**570** MVARs" to clearly be under the acceptable curve. - rlc.

NRC K/A System/E/A System 2.1 Conduct of Operations

Generic K/A Selected

NRC K/A Generic System 2.1 Conduct of Operations

Number 2.1.25 **RO** 3.9 **SRO** 4.2 **CFR Link** (CFR: 41.10 / 43.5 / 45.12)

Ability to interpret reference materials, such as graphs, curves, tables, etc.

Question #: N/A | Question ID: 78685 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☐ Past NRC Exam?

PARENT

AOP 2580, Degraded Voltage, has the crew refer to Attachment 1, "Estimated Capability Curves" and ensure operation is within limits.
The main generator output is currently 800 MWe and hydrogen pressure is 60 psig.

What is the maximum MVARs lagging that the generator can produce and stay within the limits of the curve?

☐ **A** 380 MVARs

☐ **B** 440 MVARs

☑ **C** 520 MVARs

☐ **D** 620 MVARs

Justification

AOP 2580, Attachment 1, at 800 MWe and 60# hydrogen, the max MVAR loading is 520 MVARs. This questionn requires reference to AOP 2580, Attachment 1.

References

AOP-2580, Att. 1 Curve

NO Comments or Question Modification History at this time.

NRC K/A System/E/A System

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

Question #: 69 | Question ID: 1000034 | ☑ RO | ☐ SRO | ☑ Student Handout? | ☐ Lower Order?
Rev. 1 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

Surveillance procedure 2612A 'A' Service Water Pump Tests is being performed to verify that the pump is capable of generating acceptable differential pressure.

A PEO in the intake structure will measure the 'Distance from floor to Circ Water Bay level' and read the 'Discharge pressure' from the strainer inlet. He will then report these values to the Control Room.

Which of the following sets of data will meet the Acceptance Criteria?

☐ **A** 10,250 gpm header flow, 46.3 psig discharge pressure, 7 feet from floor to water level

☑ **B** 10,350 gpm header flow, 38.9 psig discharge pressure, 15 feet from floor to water level

☐ **C** 10,550 gpm header flow, 40.1 psig discharge pressure, 8 feet from floor to water level

☐ **D** 10,650 gpm header flow, 46.4 psig discharge pressure, 12 feet from floor to water level

Justification

B: correct, although discharge pressure is below the line the large distance to the water level indicates a very low tide, since the required values are referenced to a mean sea level (14') the lower suction head translates to a delta-P of 39.35 psid.
14' - 15' = -1'; -1 x 0.45 = -0.45; 38.9 - (-0.45) = 39.35

A: wrong; Minimum acceptable flow rate is 10,300 gpm.
Plausible; Conditions result in 43 psid pressure, which is well within acceptable margin (38.6 - 45.1 psid).

C: wrong; corrected value is 37.4 psid, which is below the acceptable range of 38.6.
Plausible; With a higher flow rate and discharge pressure than "B", this set looks acceptable.

D: wrong; Corrects to 45.5 psid, which is above the range for acceptable differential pressure.
Plausible; Flow rate and corrected pressure are very good, but pressure is too good.

References **Provided**

Requires use of form SP-2612A-003, R3C0

Comments and Question Modification History

02/01/11; Per validation, modified answer (changed "**14**" to "**15**" feet) to be within "Normal" limits and corrected math error in Justification. - rlc.

NRC K/A System/E/A **System** 2.2 Equipment Control

Generic K/A Selected

NRC K/A Generic **System** 2.2 Equipment Control

Number 2.2.12 **RO** 3.7 **SRO** 4.1 **CFR Link** (CFR: 41.10 / 45.13)
Knowledge of surveillance procedures.

Question #: 70 | Question ID: 1154135 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The plant is in Mode 5, making preparations to come out of a refueling outage. Two operators have been instructed to perform an Independent Verification of the RBCCW system valve alignment inside containment. The first operator finds the 'A' CEDM Cooler Outlet Throttle Valve, 2-RB-35A, open but UNLOCKED. The Shift Manager has given direction for the valve to be repositioned to 1 turn open and locked, per the valve alignment sheet.

Which of the following describes the actions necessary to position and lock 2-RB-35A as instructed?

☑ **A** A second operator will verify the first operator fully closes the valve, then reopens the valve to one full turn open and locks it in that position. Then a third operator, BY HIMSELF, will verify the valve is locked.

☐ **B** A second operator will verify the first operator fully opens the valve, then closes the valve the same number of turns and locks it in that position. Then a third operator, BY HIMSELF, will verify the valve is locked.

☐ **C** One operator, BY HIMSELF, will fully close the valve, reopen it one full turn and lock it in that position. Then, the second operator will verify the valve is open and locked by visual inspection of the valve.

☐ **D** One operator, BY HIMSELF, will fully close the valve, reopen it one full turn. Then, the second operator will verify the valve is properly positioned using system parameters, then lock it in that position.

Justification

A - CORRECT; PI-AA-500, describes the requirements for Independent and Concurrent Verification. Attachment 2 specifies that Concurrent Verification is to be used for positioning a throttle valve that is required to be verified.

B - WRONG; Throttle valves are verified by fully closing then reopening to position, if system operation allows it. Based on stem information, this method should be used.
Plausible; This method would be acceptable for throttling a valve that is normally full open, to some new desired position, especially if totally stopping flow was unacceptable.

C - WRONG; Re-positioning a throttle valve requires Concurrent Verification and this does not meet that criteria.
Plausible; This is the acceptable method for all other mechanical valves.

D - WRONG; The valve must immediately be locked in position before the operator positioning it leaves.
Plausible; This is the acceptable method if the throttle valve was not a "locked" valve.

References

PI-AA-500, R1, Attachment 2, Pg. 12 of 14

Comments and Question Modification History

02/02/11; Per validation, added **"Then a third operator, BY HIMSELF, will verify the valve is locked."** to correct answer to make it correct. Also modified distracter 'B' to balance the change in choice 'A'. - rlc.

NRC K/A System/E/A **System** 2.2 Equipment Control

Generic K/A Selected

NRC K/A Generic **System** 2.2 Equipment Control

Number 2.2.14 **RO** 3.9 **SRO** 4.3 **CFR Link** (CFR: 41.10 / 43.3 / 45.13)
Knowledge of the process for controlling equipment configuration or status.

Question #: N/A | Question ID: 54135 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☐ Past NRC Exam?

PARENT

While performing the RBCCW valve lineup, the 'A' CEDM Cooler Outlet Throttle Valve, 2-RB-35A, is found open but UNLOCKED. The Shift Manager has given direction for the valve to be repositioned to 1 turn open and locked, per the valve alignment sheet.

Which of the following statements describes the actions to be taken before 2-RB-35A can be signed off as throttled to its proper position?

- ☐ **A** One operator, BY HIMSELF, will fully open the valve, counting the number of turns, then close the valve the specified number of turns to ensure the desired result of the controlled parameter is achieved.
- ☐ **B** One operator, BY HIMSELF, will count the number of turns required to close the valve and then open the valve the same number of turns after verifying the number of turns closed was correct.
- ☐ **C** A second operator will independently verify that the valve is properly positioned by observing stem position and verifying the desired result of the controlled parameter is achieved.
- ☑ **D** A second operator will verify the valve is fully closed and then observe the first operator open the valve 1 full turn.

Justification

WC-6 Attachment 2 Defines Dual Verification. Attachment 3 specifies that Dual Verification is to be used for positioning a throttle valve that is required to be verified, Which is also required by Attachment 2. Attachment 4 specifies the method.

References

NO Comments or Question Modification History at this time.

NRC K/A System/E/A System

Number	RO	SRO	CFR Link

Question #: 71 | Question ID: 1130001 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 1 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The following plant conditions exist:

* 100% power
* All systems are in normal operation
* Forcing pressurizer (PZR) Sprays for boron equalization

Then, a 480 VAC bus 22A deenergizes on a bus fault, resulting in several component failures, including the following:

- Group 1 PZR Backup Heater trips causing RCS pressure to stabilize at 2220 psia.
- The Turbine Battery charger deenergizes causing the battery to begin discharging.
- VR-21 UPS alarms on AC supply power input voltage low.

Which of the following items requires entry into a Technical Specification Action Statement, based on the above plant conditions?

☑ **A** RCS pressure

☐ **B** Backup heaters

☐ **C** VR-21 UPS

☐ **D** VA-40 Alternate Supply

Justification

A; CORRECT - RCS pressure has dropped below the DNB minimum requirements of 2225 psia per Tech. Specs.

B; WRONG - Only the Proportional Heaters are covered by Tech. Specs.
Plausible; Tech. Specs. do require at least two groups of heaters be operable, but they must be vitally powered.

C; WRONG - The VR-21 UPS is a backup to a Non-Vital Instrument AC source that is not covered by Tech. Specs.
Plausible; VR-21 UPS can be powered in such a way that it will make the EDG inoperable. Also, it is the backup power supply to Vital Instrument AC panel VA-40, which is covered by Tech. Specs.

D; WRONG - Although the "backup" power supply to a Tech. Spec. required vital power supply (VA-40) has been lost, this does not, in and of itself, make the applicable vital power supply, VA-40, inoperable.
Plausible; The loss of support systems for other Tech. Spec. covered components would require entry into the applicable TSAS.

References

1. Tech. Specs. 3.2.6 "DNB Margin"
2. TRM 8.1-3, 2.7 DNB Margin, Item "b."

Comments and Question Modification History

02/01/11; Per validation, revised question to solicit Tech. Spec. applicability of new VR-21 UPS and impact on VA-40 alternate supply, instead of just VR-21. - rlc.

NRC K/A System/E/A System 027 Pressurizer Pressure Control System (PZR PCS) Malfunction

Generic K/A Selected

NRC K/A Generic System 2.2 Equipment Control

Number 2.2.22 **RO** 4.0 **SRO** 4.7 **CFR Link** (CFR: 41.5 / 43.2 / 45.2)

Knowledge of limiting conditions for operations and safety limits.

Question #: N/A | Question ID: 3000001 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☐ Past NRC Exam?

PARENT

The following plant conditions exist:

* 100% power
* All systems are in normal operation
* Forcing PZR Sprays for boron equalization

Then, VR-21 is lost causing all Backup Heaters to trip. RCS pressure drops to 1995 psia before turning due to manipulations of the remaining available pressure control components. VR-21 remains deenergized.

Which of the following describes an applicable action for the above plant conditions?

☑ **A** Restore RCS pressure to within its limits within two (2) hours.

☐ **B** Restore at least two (2) groups of Backup heaters within 72 hours.

☐ **C** Be in HOT STANDBY within one (1) hour.

☐ **D** Restore the inoperable bus to operable status within eight (8) hours.

Justification

A; CORRECT - RCS pressure has dropped below the DNB minimum requirements and must be restored within 2 hours.
B; WRONG - 3.4.4 requires at least two groups of heaters, but they must be vitally powered. Therefore, only the Proportional Heaters are applicable to this Tech. Spec.
C; WRONG - Although the pressure is below one of the minimum pressure lines on the Safety Tech. Spec., the corresponding power does not align properly with this minimum pressure on the "unacceptable" side of the line.
D; WRONG - Although the "backup" power supply to a Tech. Spec. required vital power supply has been lost, this does not, in and of itself, make the applicable vital power supply inoperable.
Requires the use of Technical Specifications

References

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 027 Pressurizer Pressure Control System (PZR PCS) Malfunction

Generic K/A Selected

NRC K/A Generic **System** 2.2 Equipment Control

Number 2.2.22 **RO** 4.0 **SRO** 4.7 **CFR Link** (CFR: 41.5 / 43.2 / 45.2)

Knowledge of limiting conditions for operations and safety limits.

Question #: 72 | Question ID: 55154 | ☑ RO ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 1 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

Which of the following describes why detectors operating in the Limited Proportional Region are NOT used as area radiation monitors at Millstone Unit 2?

☐ **A** Many ion pairs rejoin prior to reaching the anode and cathode; therefore, they are NOT counted.

☐ **B** Because of the avalanche effect, only pulses are counted, NOT radiation levels.

☐ **C** The voltage is so high in this region that recombination of the formed ion pairs is NOT possible.

☑ **D** The secondary ionizations in this region are erratic, even with a stable voltage.

Justification

In this region, the Gas Amplification process is still occurring, but is no longer constant with respect to voltage. The number of secondary ionizations are limited by the slow-moving positive ions near the anode.

A is incorrect. This describes the Recombination Region, which is also NOT suitable for Area Radiation Monitoring.
Plausible: The examinee may remember that the description of this type of detector is NOT suitable for Area Radiation Monitoring, but may NOT remember the description for the Limited Proportional Range.

B is incorrect. This describes the Geiger-Mueller Region, which may be used for personal or equipment contamination, but is also NOT suited for determining area radiation levels.
Plausible: The examinee may remember that the description of this type of detector is NOT suitable for Area Radiation Monitoring, but may NOT remember the description for the Limited Proportional Range.

C is incorrect. This describes the Continuous Discharge Region, which is also NOT suited for Area Radiation Monitoring.
Plausible: The examinee may remember that the description of this type of detector is NOT suitable for Area Radiation Monitoring, but may NOT remember the description for the Limited Proportional Range.

References

RMS-00-C, Radiation Monitoring System

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.3 Radiation Control

Generic K/A Selected

NRC K/A Generic **System** 2.3 Radiation Control

Number 2.3.15 **RO** 2.9 **SRO** 3.1 **CFR Link** (CFR: 41.12 / 43.4 / 45.9)

Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Question #: **73** | **Question ID:** 1150023 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | **Origin:** Mod | ☐ Past NRC Exam?

The plant tripped from 100% power due to a rupture of an RCP Seal Bleedoff tube inside the RCP seal cooler.
EOP-2525, Standard Post Trip Actions have just been completed and the following conditions now exist:

- The pressurizer is at 7% and slowly rising
- RBCCW Surge Tank is at 99% and rising
- RWST level is 98% and slowly lowering
- SIAS, CIAS and EBFAS have been fully verified
- No manual operator actions have yet been taken

Which of the following describe additional radiological hazards, based on the above conditions?

☐ **A** Increased Enclosure Building ventilation flow due to EBFAS.

☐ **B** Increased Charging flow in the -5 ft. West Piping Penetration area.

☐ **C** Starting of LPSI and HPSI pumps in the -45 ft. level Aux. Building.

☑ **D** Overflow of the RBCCW Surge Tank in the Enclosure Building.

Justification

D - CORRECT; The tube leak in the RCP Seal Cooler is a direct flow path from the RCS to RBCCW; therefore, the RBCCW Surge Tank will fill up and overflow to the East 38'6" Penetration Room resulting in a radioactive release to the Enclosure Building.

A - WRONG; The increase ventilation flow in the Enclosure building is NOT a radiological concern due to the filtration of the air flowing through the Enclosure building Filtration System.
Plausible: The examinee may feel that the radiological release in the Enclosure Building from the RBCCW Surge Tank will be worse (more wide spread) with the increase in air flow from the EBFAS Fans running.

B - WRONG; Starting additional charging Pumps will NOT result in increased radiation readings locally because the Charging Pumps are taking a suction from either the RWST or the Boric Acid Storage Tanks which have very low (insignificant) radiation levels.
Plausible: During normal operation, starting additional Charging Pumps results in a rise in local radiation levels due to the associated increase in Letdown flow. In this event, Letdown is isolated; therefore, additional Charging does NOT raise Letdown flow.

C - WRONG; Starting the Safety Injection Pumps is NOT likely to cause any difference in radiation levels because flow would be from the RWST, which has extremely low (insignificant) radiation levels.
Plausible: The examinee may feel that starting Safety Injection Pumps will result in elevated radiation levels in the Aux and Enclosure buildings.

References

1. EOP-2532, R29C1; Step 7
2. RBC-00-C, R5, Section F "Malfunctions and Failures", Pg. 43 - RBCCW and Pg. 48 - PMW System failures.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.3 Radiation Control

Generic K/A Selected

NRC K/A Generic **System** 2.3 Radiation Control

Number 2.3.14 **RO** 3.4 **SRO** 3.8 **CFR Link** (CFR: 41.12 / 43.4 / 45.10)

Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Question #: N/A | Question ID: 5000023 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☑ Past NRC Exam?

PARENT

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)

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.3 Radiation Control

Generic K/A Selected

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.

Question #: **74** | **Question ID: 56128** | ☑ RO ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 1 | ☑ Selected for Exam | **Origin: Bank** | ☐ Past NRC Exam?

AOP 2579A and AOP 2579AA are both Fire Procedures for Appendix R Fire Area R-1.

What is the difference between the two procedures?

☐ **A** AOP 2579A deals with shutting down and cooling down the plant, while AOP 2579AA deals with the equipment damaged by the fire.

☑ **B** AOP 2579A deals with getting the plant to Hot Standby, while AOP 2579AA deals with getting the plant to Cold Shutdown.

☐ **C** AOP 2579A deals with the actions performed at the Fire Shutdown Panel (C10), while AOP 2579AA deals with the actions performed locally in the plant.

☐ **D** AOP 2579A deals with placing the plant in a stable condition, while AOP 2579AA deals with actual fire fighting requirements.

Justification

B - CORRECT; AOP 2579A, Fire Procedures for Hot Standby Appendix R Fire Area R-1, deals with getting the plant to Hot Standby, while AOP 2579AA, Fire Procedures for Cool Cold Shutdown Appendix R Fire Area R-1, deals with getting the plant to Cold Shutdown.

"A" is incorrect because 2579A does NOT provide instructions for a plant cooldown and 2579AA does NOT provide guidance for equipment damage. AOP 2579A only provides guidance for Hot Standby. Maintenance procedures will provide guidance on equipment damaged by the fire.
Plausible: The examinee may NOT be familiar with the specifics of that is contained in the Appendix 'R' fire procedures.

"C" is incorrect because 2579A and 2579AA do NOT differentiate between actions performed in the plant vs. actions performed at C-10. They both provide direction for C-10 and in-plant.
Plausible: The examinee may NOT be familiar with the specifics of that is contained in the Appendix 'R' fire procedures.

"D" is incorrect because 2579AA does NOT provide any direction for fire fighting. Fire fighting strategies are provided by the fire brigade procedures.
Plausible: The examinee may NOT be familiar with the specifics of that is contained in the Appendix 'R' fire procedures.

References

AOP-2579A and AOP-2579AA, Purpose

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.4 Emergency Procedure /Plan

Generic K/A Selected

NRC K/A Generic **System** 2.4 Emergency Procedures /Plan

Number 2.4.25 **RO** 3.3 **SRO** 3.7 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of fire protection procedures.

Question #: 75 | Question ID: 1100033 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant has tripped from 100% power due to a loss of DC bus 201B (Battery bus breaker trip).
The following additional conditions exist:

- On the loss of bus 201B, the "A" Main Steam header ruptured in containment.
- Bus 24C is being power by the "A" Emergency Diesel Generator.
- All other components are functioning as designed based on the above casualties.
- The crew is performing the actions of EOP-2525, Standard Post Trip Actions.

Which one of the following local actions are required and why?

☐ **A** Trip the "B" Aux. Feedwater pump breaker to prevent feeding the affected SG.

☐ **B** Operate the Turbine Driven Aux. Feedwater Pump to control #2 SG level.

☑ **C** Cross-tie Station Air with Unit 3 to allow for remote ADV operation to control RCS temperature.

☐ **D** Operate the "B" Atmospheric Dump Valve remotely from C-21 to control RCS temperature.

Justification

C - CORRECT; The loss DV-20 will cause 24D to de-energize on the subsequent plant trip. The "D" IAC lost power when 24C did not transfer to the RSST and was pick up by the EDG (LOOP). On a Loss Of Offsite Power with a concurrent SIAS, the operators are not allowed to re-start the vital IAC and are required to cross-tie air with Unit 3.

A - WRONG; Control power to this valve (VA-20) is normally from DC bus 201B, VA-20 would still be energized by its alternate power supply, INV-6 (Turbine Battery).
Plausible; Loss of 201B de-energizes half of the vital DC busses and, if not for the alternate power supply to VA-20, would require local AFRV control.

B - WRONG; The BOP can swap control power for the TDAFP to DV-10 using the key switches on C05, and use it to supply AFW.
Plausible; DV-20, the normal supply to the TDAFP, was lost with the loss of 201B. Loss of control power would require use of a PEO.

D - WRONG; Control of the "B" ADV from C-05 was not lost because VR-21 is still energized by the new UPS, which is good for one to four hours.
Plausible; In the recent past, loss of 24D would cause a loss of VR-21. After about 10 minutes, the battery backup for Foxboro IA control signals (normally powered by VR-21) would deplete and prevent control of the "B" ADV from the control room.

References

AOP 2504B, R3C11, Pg 4, Discussion Section

Comments and Question Modification History

01/06/11; Modified stem to state that EOP-2525 actions are in progress, not completed, per comments from Sandy Doboe. - rlc

02/01/11; Per validation, changed choice "A" from "control B Aux. Feedwater Reg. valve" to "trip "B" Aux Feed pump breaker" due to loss of DC possible effect on AFRV control circuit. - rlc

NRC K/A System/E/A **System** 2.4 Emergency Procedure /Plan

Generic K/A Selected

NRC K/A Generic **System** 2.4 Emergency Procedures /Plan

Number 2.4.35 **RO** 3.8 **SRO** 4.0 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

Question #: 76 | Question ID: 1100034 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant has tripped from 100% power due to a malfunction in the Turbine Control System.
The following plant conditions now exist:

- US directs the performance of EOP 2525, Standard Post Trip Actions.
- Two CEAs are stuck out.
- Bus 24C is faulted.
- Facility 2 SIAS, CIAS, EBFAS, CSAS, and MSI verified fully actuated with all components functioning as designed.
- All other electrical buses are energized.
- NO RCPs are operating.
- Pressurizer level is 0% and NOT restoring.
- Reactor Vessel Head level is 43% and stable.
- All available charging pumps are operating, but charging flow is 'zero'.
- Pressurizer pressure is 350 psia, and continuing to lower.
- SG levels are 45% and stable.
- SG pressures are 785 psia and stable.
- CETs are 432° F and stable.
- Containment pressure is 19 psig and slowly rising.
- Containment temperature is 143°F and slowly rising.
- Containment high range radiation monitors indicate 0.01 R/hr and stable.
- Steam plant radiation monitors are NOT in alarm, NOT going up.
- Radiation monitors outside Containment are NOT in alarm, NOT going up.

Then, at the completion of EOP-2525, while the US is evaluating Contingency Actions taken, DC bus 201B deenergizes.

Which of the following actions must the US perform?

☐ **A** Determine Inventory Control is NOT met, skip the Diagnostic Flow Chart, and immediately transition to EOP 2540C1, Functional Recovery of Inventory Control.

☐ **B** Determine Reactivity Control is NOT met, skip the Diagnostic Flow Chart, and immediately transition to EOP 2540A, Functional Recovery of Reactivity Control.

☐ **C** Determine from the Diagnostic Flow Chart that Pressure Control is NOT met and immediately transition to EOP 2532, Loss of Coolant Accident.

☑ **D** Determine from the Diagnostic Flow Chart that Vital Auxiliary Control is NOT met and immediately transition to EOP 2540, Functional Recovery.

Justification

D is correct. A loss of Bus 24C and Bus 201B will result in a loss of one vital AC and DC in opposite facilities resulting in a loss of Vital Auxiliaries. When using the Diagnostic Flow Chart, if the Vital Auxiliaries Safety Function is NOT met, then the flow chart directs the user to transition directly to EOP 2540, Functional Recovery.

A is incorrect. Even though Inventory Control appears to be in jeopardy, the US CANNOT skip the Diagnostic flow Chart.
Plausible: Reanalysis of LOCAs has determined that Charging is NO longer necessary. The examinee may not remember this and determine that Inventory Control is NOT met, requiring the use of EOP 2540C1 after using EOP 2540 to diagnose which Safety Function needs attention first.

B is incorrect. Even though Reactivity Control is the highest safety function, the US CANNOT skip the Diagnostic Flow Chart. Additionally Reactivity Control is being addressed by Boration with Safety Injection, NOT Charging.
Plausible: Reactivity Control is in jeopardy due to the two stuck CEAs. Because Reactivity Control is the highest safety function, the examinee may believe that it should be addressed immediately.

C is incorrect. If this were just a Large Break LOCA, then EOP 2532 would be appropriate; however, Vital Auxiliaries is affected requiring the crew to address this Safety Function through the Functional Recovery procedure.
Plausible: Pressure Control is in jeopardy due to the Large Break LOCA. If the examinee did NOT recognize the loss of a Vital AC and Vital DC Bus in opposite facilities, then this would be the appropriate response.

References

OP 2260, EOP Users Guide

Comments and Question Modification History

02/02/11; Per validation, modified first paragraph of stem to shorten. Reworded the verification of ESAS actuation and when the loss of 201B occurs to clarify the time line. - rlc

Question #: **76** | **Question ID:** 1100034 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☑ Selected for Exam | **Origin:** New | ☐ Past NRC Exam?

NRC K/A System/E/A **System** 2.4 Emergency Procedure /Plan

Generic K/A Selected

NRC K/A Generic **System** 2.4 Emergency Procedures /Plan

Number 2.4.6 **RO** 3.7 **SRO** 4.7 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of EOP mitigation strategies.

Question #: 77 | Question ID: 1100035 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant is in Mode 4, starting up from a refueling outage. "A" and "B" RCPs are running with "C" and "D" RCP breakers racked up. SDC has been terminated and the Safety Injection Systems are being returned to normal.

Then, "A" RBCCW Header flow is lost when the "A" RBCCW Heat Exchanger outlet valve fails closed. While making preparations to restore "A" RBCCW Header flow, the following indications are seen for the "A" RCP:

- RCP RBCCW Outlet temperature - 122 °F and rising slowly
- Motor Stator Temperature - 270°F and slowly rising

Which of the following directions should the US give in response to the above conditions?

☐ **A** Immediately direct the "A" RCP be tripped due to RCP RBCCW Outlet Temperature exceeding the limit, and start the "D" RCP.

☐ **B** Once RBCCW flow has been restored, direct the "A" RCP be tripped due to RCP RBCCW Outlet Temperature exceeding the limit, and start the "C" RCP.

☑ **C** Immediately direct the "A" RCP be tripped due to the Motor Stator Temperature exceeding the limit, and start the "D" RCP.

☐ **D** Once RBCCW flow has been restored, direct the "A" RCP be tripped due to the Motor Stator Temperature exceeding the limit, and start the "C" RCP.

Justification

C - CORRECT: AOP-2564, Loss Of RBCCW, gives parameters to be monitored, and associated contingency actions required, if a parameter (temperature) is exceeded based on the loss of cooling water. The stator temperature limit is 260°F, requiring "A" RCP be immediately secured. However, MP2 is not allowed to operate with just one RCP, so a second must be immediately started.

A - WRONG: RCP RBCCW Outlet temperature does NOT exceed the limit for a pump trip.
Plausible; The RBCCW HX outlet limit for RBCCW system operability is 120°F.

B - WRONG: Pump trip should not be delayed until RBCCW temperatures are restored.
Plausible; The RBCCW HX outlet limit for RBCCW system operability is 120°F and "C" RCP is on the same facility as "A" RCP. Starting "C" RCP would balance flow and heat load.

D - WRONG: Pump trip should not be delayed due to the Motor Stator temperature exceeding the limit for a pump trip.
Plausible; MP2 is not allowed to operate just one RCP and "C" RCP is on the same facility as "A" RCP. Starting "C" RCP would balance flow and heat load.

References

1. ARP-2590B-066, "RCP A STR TEMP HI", Rev. 000, Alarm setpoint is 260°F. Procedure requires a pump trip above 260°F.
2. AOP-2564, Loss Of RBCCW, step 3.3, bullet #6, Page 7 of 46, parameters to be monitored, and associated contingency actions required, on RCP high temp due to RBCCW loss.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 015 Reactor Coolant Pump Malfunctions

Number AA2.09 **RO** 3.4 **SRO** 3.5 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): When to secure RCPs on high stator temperatures

Question #: 78 | Question ID: 1190004 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The plant was manually tripped from 100% power due to a Steam Generator Tube Rupture (SGTR) on #2 SG.

The following conditions now exist:

- On the trip, 24D de-energized due to a bus fault.
- 24C/24E is energized
- SIAS, CIAS and EBFAS have actuated.
- All other plant systems respond as designed.

Which of the following actions must a PEO be directed to perform during EOP 2534, Steam Generator Tube Rupture, and what is the reason for this action?

☑ **A** Manually close #2 S/G Steam Supply to the Turbine Driven Auxiliary Feedwater Pump, MS-202, and prevent the loss of a barrier and escalation of the event classification.

☐ **B** Manually open all de-energized Facility 2 Safety Injection Valves to raise Safety Injection flow above the minimum required for RCS Inventory Control.

☐ **C** Isolate Hotwell Reject Valve to stop the potential overflow of the Condensate Surge Tank, and prevent the loss of a barrier and escalation of the event classification.

☐ **D** Locally transfer 24E to 24D to allow Facility 2 power restoration from Unit 3 and the increase of Safety Injection flow above the minimum required for RCS Inventory Control.

Justification

A - CORRECT; A SGTR on #2 S/G requires the associated side steam supply to the TDAFP to be closed which will prevent the unmonitored release of radioactivity from the TDAFP exhaust. The #2 S/G Steam Supply to the TDAFP, MS-202, must be manually closed due to the loss of power to the motor operator (Loss of B62 due to the loss of 24D).

B - WRONG; The deenergized Safety injection Valves do not need to be manually opened because they have on running pump. When Bus 24D is reenergized the Facility 2 SI Pumps and associated SI valves will be powered again.
Plausible: The Facility 2 SI LPSI injection valves will be closed and without power (loss of B61). The examinee may think that the closed injection valves need to be manually opened prior to energizing the vital bus.

C - WRONG; The loss of bus 24D does not result in an overflow of the Condensate Surge Tank.
Plausible: If the examinee believes that the loss of Bus 24D will cause the hotwell to continue to fill and reject to the Condensate Surge Tank (steam continues to dump to the Condenser with no Main Feed), then reject must be isolated.

D - WRONG; Bus 24D cannot be restored until the fault is cleared; therefore Bus 24E will not be transferred to Bus 24D.
Plausible: The examinee may believe that it is prudent to restore Bus 24D to ensure both facilities of Safety injection are in operation as soon as possible; therefore, Bus 24E would need to be manually transferred to Bus 24D.

References

EOP 2525, AOP 2504D.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A | **System** 038 | Steam Generator Tube Rupture (SGTR)

Generic K/A Selected

NRC K/A Generic | **System** 2.4 | Emergency Procedures /Plan

Number 2.4.35 | **RO** 3.8 | **SRO** 4.0 | **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

Question #: N/A | Question ID: 9000004 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☐ Selected for Exam | Origin: Parent | ☑ Past NRC Exam?

PARENT

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

Then the following events occur:

* Automatic plant trip due to a Steam Generator Tube Rupture (SGTR) on #2 Steam Generator (SG).
* Loss of the RSST and VA-20 at the time of trip.
* Shortly after the trip, a Safety Injection Actuation signal (SIAS) automatically actuated.
* All other plant systems respond as designed.

Which one of the following actions must the US perform during EOP 2525, Standard Post Trip Actions, to mitigate the consequences of this event and what is the reason for this action?

☐ **A** Dispatch a PEO to the Hot Shutdown Panel, C-21, to throttle open the #2 Atmospheric Dump Valve. This will permit a cooldown of both Hot Leg temperatures to ≤ 515°F.

☐ **B** Direct the BOP to swap the control power supply switch for the Terry Turbine to Facility 1. This will allow the operator to maintain both S/G levels in the prescribed bands.

☑ **C** Dispatch a PEO to manually operate the "B" Auxiliary Feedwater Regulating Valve, 2-FW-43B. This will prevent excessive auxiliary feedwater from overfilling the affected SG.

☐ **D** Direct the BOP to close #2 S/G Steam Supply to the Terry Turbine, MS-202, after the disconnect is closed. This will minimize the radioactive release from the affected SG.

Justification

C IS CORRECT; On a loss of normal power, Condensate is lost; therefore, main Feedwater is lost. The loss of VA-20 will cause the "B" Aux Feed Regulating Valve to fail open. EOP 2525 requires at least two Auxiliary Feed Pumps to be started. In order to prevent overfeeding #2 SG, the "B" Aux Feed Regulating Valve, 2-FW-43B, must be either closed or isolated locally.

A is incorrect; A loss of VA-20 will result in a loss of power to the #2 ADV from ALL remote locations. The #2 ADV can ONLY be operated locally with the handwheel in manual. (A loss of VR-21 will result in the loss of control to the #2 ADV from C-05. The ADV may then be controlled from C-21.)
Plausible because the examinee may think that the Facility 2 components controlled from Hot Shutdown Panel, C-21 are powered from VR-21 or VA-40 and are NOT affected by a loss of VA-20.

"B" is incorrect; Control power supply to the Turbine Driven Auxiliary Feedwater Pump is from DV-20, NOT VA-20; therefore, swapping power supplies will have NO impact on the availability of the Turbine driven Auxiliary Feedwater Pump.
Plausible because the examinee may not remember that the power supply for the TDAFP is DV-20 NOT VA-20

"D" is incorrect; #2 S/G Steam Supply to the Terry Turbine, MS-202, will be closed to minimize the release of radioactive steam from the Terry Turbine exhaust; however, this action CANNOT be performed in EOP 2525. This action is only performed in EOP 2534, after lowering both hot leg temperatures to <515°F, when isolating the affected S/G.
Plausible because this action will be performed at a later time and for the stated reason.

References

EOP 2525, AOP 2504D.

Comments and Question Modification History

Bob K. - D-4/C
Bill M. - D-2/C, K
Angelo - D-4/C; **Change "close" to "operate" in correct answer. - RLC**
Bruce F. - D-4/C, No comment

NRC K/A System/E/A System 038 Steam Generator Tube Rupture (SGTR)

Generic K/A Selected

NRC K/A Generic System 2.1 Conduct of Operations

Number 2.1.30 **RO** 4.4 **SRO** 4.0 **CFR Link** (CFR: 41.7 / 45.7)
Ability to locate and operate components, including local controls.

Question #: **79** | **Question ID:** 1100037 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | **Origin:** New | ☐ Past NRC Exam?

The plant is in MODE 2 in preparation for warming the Main Turbine. The "A" Main Feed Pump is in service supplying both Steam Generators.

Suddenly, the following indications are received and reported:

- Numerous alarms are annunciated on C-08
- #2 FRV Bypass Valve is closed.
- RCS temperature is 531°F and slowly rising.
- RCS pressure is 2235 psia and lowering.

The US directs the RO and BOP to stabilize the plant per the appropriate AOP.

What caused this transient and which of the administrative requirements must be followed?

☐ **A** Vital DC Bus, DV-20, is deenergized.
Restore the inoperable bus to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

☐ **B** Vital Instrument AC Bus, VA-20, is deenergized.
Restore the inoperable inverter to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

☑ **C** Vital Instrument AC Bus, VA-20, is deenergized.
Restore the inoperable bus to OPERABLE status within 8 hours or be in COLD SHUTDOWN within the next 36 hours.

☐ **D** Vital DC Bus, DV-20, is deenergized.
Restore the inoperable bus to OPERABLE status within 8 hours or be in COLD SHUTDOWN within the next 36 hours.

Justification

C is correct. Examinee must determine from the given indications that VA-20 is lost then determine the appropriate action required by Tech Specs.

A is incorrect. The indications do NOT necessarily indicate that DC bus, DV-20, is deenergized. Additionally, DV-20 is NOT required to be energized to consider the Vital DC bus OPERABLE.
Plausible: Loss of DV-20 would cause other similar responses such as a the Facility 2 D/G will start, the #2 MSIV will fail closed, Letdown will isolate, etc. The examinee may not remember which components change state on a loss of a Vital AC Instrument bus. The Tech Spec Action is plausible in that it is correct for a loss of Vital DC Bus, 201B, which powers DV-20.

B is incorrect. While a malfunction on Inverter 2 will cause VA-20 to shift to Inverter 6, the indications reflect a loss of VA-20, NOT just a failure of Inverter 2.
Plausible: The examinee may believe that the VA-20 swap from Inverter 2 to Inverter 6 is a break-before-make transfer and that momentarily deenergized components must be reset to restore proper operation. The Tech Spec Action is correct for an inoperable Inverter.

D is incorrect. Although indications may be similar, the given indications do NOT reflect a loss of Vital Bus 22F.
Plausible: A loss of Bus 22F will cause a loss of Non-Vital Instrument Bus VR-21. A loss of VR-21 will result in the loss of numerous indications and controls important for operation. The examinee may NOT remember which indications are lost for a loss of either bus. The Tech Spec Action is correct for a loss of Bus 22F.

References

AOP 2540D
TS 3.8.2.1

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 057 Loss of Vital AC Electrical Instrument Bus

Generic K/A Selected

NRC K/A Generic **System** 2.4 Emergency Procedures /Plan

Number 2.4.47 **RO** 4.2 **SRO** 4.2 **CFR Link** (CFR: 41.10,43.5 / 45.12)
Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Question #: 80 | Question ID: 1100038 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant is operating normally in MODE 1 at 100% power with the following conditions:
- Average injection temperature over the past 24 hours is 74.7°F
- "A" and "C" Service Water Pumps are supplying Facility 1 and 2, respectively.
- Bus 24E is aligned to Bus 24D.

The "A" Service Water Pump suddenly trips on overload. The BOP successfully starts the "B" Service Water Pump on Facility 1 and completes the actions of AOP 2565, Loss of Service Water. A PEO has just informed the US that the injection temperature has slowly risen over the past 24 hours and is now at 75.2° F.

Which of the following describes the administrative actions for these conditions?

☐ **A** With both Service Water headers inoperable, log into Tech Spec 3.0.3. Restore either Header to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours.

☑ **B** With "A" Service Water Header inoperable, log into Tech Spec 3.7.4.1. Restore the header to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 36 hours.

☐ **C** With the Ultimate Heat Sink above the maximum allowed value, log into Tech Spec 3.7.11. Be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours.

☐ **D** With both Service Water Headers inoperable, log into Tech Spec 3.0.3 until the "B" Service Water Strainer is vented, then log into Tech Sec 3.7.4.1 until the "A" header is restored.

Justification

B is correct. When the "A" Service Water Pumps trips, the "A" Service Water header is inoperable. This requires entry into Service Water Tech Spec Action 3.7.4.1 The Action Statement applies even when the "B" Service Water Pump is started on the "A" Service Water Header. The "B" Service Water Pump breaker is normally left in the "BLOCK" position to ensure it will NOT start on the "B" Diesel Generator if a subsequent SIAS or LNP were to occur.

A is incorrect. Tech Spec 3.0.3 is NOT applicable with the "B" Service Water Pump breaker in "BLOCK".
Plausible: If the examinee does NOT remember the normal position of the "B" Service Water Pump breaker "SIAS/LNP Block Switch". With the switch in "NORMAL", the "B" SW Pump would be supplying Facility 2 and would start on the "B" D/G during a SIAS or LNP.

C is incorrect. The Ultimate Heat Sink has exceeded the maximum value of 75°F, but has not exceeded the maximum value of 77°F, which would require entry into the stated action statement..
Plausible: If the Ultimate Heat sink temperature exceeds 75°F, then the average injection temperature over the past 24 hours must be verified not to exceed 77°F. If the examinee may NOT remember the actual value for entry into the action statement that requires a plant shutdown.

D is incorrect. The "B" Service Water Pump is NOT inoperable because the strainer has NOT been vented yet.
Plausible: On a normal Pump swap, the "B" Service Water Pump (and associated header) is NOT considered OPERABLE until the strainer has been vented; however, Tech Spec 3.7.4.1 is applicable while the "B" Service Water Pump is electrically aligned to Facility 2 and is NOT available to start on SIAS or LNP

References

AOP-2565, Loss of Service Water, St. 10.3
AOP-2564, Loss of RBCCW, St. 3.3.1 (Contingency)

Comments and Question Modification History

02/01/11; Per validation, changed stem to state a **PEO informs the US** of injection temperature rise over the last 24 hours, as this is how the US would normally become aware of this info. Also, changed injection temperature to what it is presently at, instead of the average, so TS 3.7.11 would not apply. - rlc.

NRC K/A System/E/A **System** 062 Loss of Nuclear Service Water

Generic K/A Selected

NRC K/A Generic **System** 2.4 Emergency Procedures /Plan

Number 2.4.11 **RO** 4.0 **SRO** 4.2 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of abnormal condition procedures.

Question #: 81 | Question ID: 1100040 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

With the plant operating at 100% power, a PEO performing a valve alignment reports a broken Instrument Air line in the Intake Structure. Another PEO is dispatched to close Intake Structure Header Isolation, 2-IA-24.

What additional action must the US direct and why?

☑ **A** Manually start screen wash pumps and Travelling Screens in FAST speed.
All screen D/P instruments fail to zero which will prevent each of the screens from automatically starting.

☐ **B** Manually start the backwashing of all Service Water Pump Strainers.
All Strainer Backwash Valves fail open resulting in all three Service Water Strainers becoming inoperable.

☐ **C** Manually close one of the Service Water Header Cross-Tie Valves, SW-97A or SW-97B.
With both Service Water Cross-Tie Valves failed open, one header must be declared inoperable.

☐ **D** Manually open the Domestic Water Supply to the Circ Water Pump Lube Water Header.
The Service Water supply to Circ Water Lube Water fails closed causing a loss of lube water flow.

Justification

A is correct. The Screen differential pressure bubblers fail to zero on a loss of Instrument Air preventing the travelling screens from automatically starting on a high D/P. The screens must be placed in Manual Fast speed to prevent an actual high differential pressure from developing across the screens.

B is incorrect. Manually starting the SW screens will avtually make them inoperable.
Plausible: The examinee may remember the screens are inoperable if not operating normally and rotating when the flush valve is open, which is what they would normally do when backwashing. However, by "manually" starting them in backwash mode they are no longer in their TS required "automatic" mode.

C is incorrect. The Service Water Header Cross-Tie Valves, SW-97A and B do NOT fail open on a loss of Instrument Air; therefore, both headers remain operable.
Plausible: The examinee may NOT remember that the fails fail as is. If both cross-tie valves are open at the same time, then one header is considered inoperable.

D is incorrect. The Service Water Supply to Circ Water Lube Water actually fails open on a loss of Instrument Air. Opening the Domestic Water Supply to Circ Water Lube Water would actually supply too much flow to the Circ Water Pump bearings.
Plausible: If the examinee thought that the Service Water Supply to Circ Water Lube Water failed closed, then Domestic Water the Circ Water Pump bearing would need Lube Water from Domestic Water.

References

AOP-2563, R9C7, Note before step 7.3

Comments and Question Modification History

02/02/11; Per validation, added **"screen wash pumps"** to correct answer and changed choice 'B' from **sodium hypochlorite release** to **inoperability of SW strainers** to make it wrong. - rlc.

NRC K/A System/E/A **System** 065 Loss of Instrument Air

Generic K/A Selected

NRC K/A Generic **System** 2.1 Conduct of Operations

Number 2.1.32 **RO** 3.8 **SRO** 4.0 **CFR Link** (CFR: 41.10 / 43.2 / 45.12)

Ability to explain and apply system limits and precautions.

Question #: 82 | Question ID: 1100039 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 1 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant was operating at 100% power when Regulating Group 7 CEA #41 slipped to 146 steps withdrawn.
Turbine load was lowered and the plant was stabilized. All other CEAs remain fully withdrawn.

I&C has completed repairs on CEA #41 control circuit and the US has directed the RO, per AOP-2556, CEA Malfunctions, to commence recovery of CEA #41.
The RO then bypasses the applicable CEDS interlock that is preventing CEA motion and begins to withdraw CEA #41.

What is the basis for the interlock that the RO must bypass to recover CEA #41?

☐ **A** The abnormal CEA alignment would cause a high calculated core flux tilt, which would result in TM/LP pretrips on RPS. The triggered interlock stops all CEA withdrawal, which has the potential to amplify the abnormal tilt beyond that assumed for the LSSS setpoint determination.

☐ **B** The CEA has slipped below the maximum allowed insertion point designed to ensure adequate Shutdown Margin, for the existing power level. The triggered interlock stops additional CEA insertion, which could degrade Shutdown Margin below that assumed in the safety analysis.

☑ **C** The abnormal CEA alignment would cause a distortion in core power distribution, resulting in potentially high localized power levels. The triggered interlock stops all CEA movement, which could amplify these distortions in core power beyond that assumed in the safety analysis.

☐ **D** The CEA has slipped below the maximum allowed insertion point designed to ensure even fuel burnup, when operating > 20% power. The triggered interlock stops further CEA insertion, which could amplify distortions in fuel burnup beyond that assumed for LSSS setpoint determination.

Justification

C - CORRECT; CEA #41 is >8 steps misaligned from another CEA in its group. Therefore, it would have triggered a CMI on Deviation Backup which prevents all further CEA motion. The basis for Tech. Spec. 3.1.3.1 states that CEA misalignment of >/= 20 steps within a group can distort power distribution beyond that assumed in the generation of the LCO and LSSS setpoints.

A - WRONG; The interlock described is a CEA Withdrawal Prohibit (CWP), which can not be bypassed by operator action.
Plausible; A dropped CEA could possibly shift ASI enough to cause TM/LP pretrips, which would then result in a CWP being triggered. Also, the function of the CWP is to prevent operators from continuing to withdrawing CEAs and making the problem worse.

B - WRONG; This statement describes the Transient Insertion Limit, or Power Dependent Insertion Limit (PDIL), which varies as a function of the highest of nuclear or delta-T power. However, at 100% power, the PDIL interlock setpoint is ~139 steps.
Plausible; The CEA is below the PDIL setpoint for the PPC, which would give numerous alarms on this condition once the operators reset pulse counts to the actual CEA position (performed after actual rod position is verified, as part of Dropped CEA recovery).

D - WRONG; This statement describes the Long Term Steady State Insertion Limit, which has no interlock function when violated.
Plausible; The CEA is below the LTSSIL for this power level and continued operation at this level will result in unanalyzed fuel burnup.

References

Tech. Spec. Bases for 3.1.3.1

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 003 Dropped Control Rod

Number AA2.04 **RO** 3.4* **SRO** 3.6* **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Dropped Control Rod: Rod motion stops due to dropped rod

Question #: 83 Question ID: 1100041 ☐ RO ☑ SRO ☐ Student Handout? ☑ Lower Order?

Rev. 0 ☑ Selected for Exam Origin: New ☐ Past NRC Exam?

A fire alarm is received for the East DC Switchgear room and a PEO is dispatched. The investigating PEO reports that a smoke detector in the room appears to have failed, but the fire suppression system otherwise appears operational.

Which of the following actions must the US now take, based on administrative requirements for a Fire Watch in the East DC Switchgear room?

☑ **A** Within one hour, establish a roving Fire Watch to patrol the room at least once per hour.

☐ **B** Repair the detector within one hour, or station a Fire Watch in the room within the next hour.

☐ **C** Establish a continuous Fire Watch to patrol the room before and after the On-Watch PEO.

☐ **D** Direct the investigating PEO to remain in the room until an hourly Fire Watch is established.

Justification

A - CORRECT; Per TRM 3.3.3.7, Action a. "Within 1 hour, establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour unless the instrument(s) is located inside the containment."

B - WRONG; Fire Watch must be established within one hour, even if efforts are underway to repair the instrument.
Plausible; Implies Tech. Spec. allowance for "return to operability" for Fire Watch requirements and need for continuous coverage once established due to safety significance of the room.

C - WRONG; Fire Watch must patrol the room every hour, regardless of On-Watch PEO status.
Plausible; One hour before and after PEO rounds would imply full coverage per the TRM.

D - WRONG; The TRM allows one hour to station a qualified Fire Watch.
Plausible; Implies immediate coverage is necessary due to safety significance of the DC rooms.

References

TRM 3.3.3.7, Action a.

Comments and Question Modification History

02/01/11; Per validation, changed choice 'C' from "**one hour before and after**" to "**continuous patrol before and after**" and changed choice 'D' from "**a roving**" to "**an hourly**". - rlc

NRC K/A System/E/A **System** 067 Plant fire on site

Number AA2.15 **RO** 2.9 **SRO** 3.9 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Plant Fire on Site: Requirements for establishing a fire watch

Question #: 84 | Question ID: 1100042 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 1 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant has tripped from 100% on a trip of the Main Turbine due to loss of off-site power. On the trip, a Pressurizer Safety Valve failed 90% open and numerous components are unavailable due to deenergized power supplies.

The crew has transitioned to EOP-2532 and is attempting to mitigate the event.

Which of the following sets of data would provide <u>definite</u> indication that the core is becoming uncovered?

☐ **A** RVLMS = 0%
CET subcooling = 0°F

☑ **B** Pressurizer pressure = 650 psia
Highest CET = 530°F

☐ **C** RVLMS = 0%
Maximum HJTC = 700° F

☐ **D** Pressurizer pressure = 450 psia
Highest CET = 456°F

Justification

B - CORRECT; RCS pressure of 650 psia and CET temperature of 530°F indicate superheat conditions at the top of the core, which is indicative of core uncovery.

A - WRONG; RVLMS at 0% means that the lowest RVLMS probe is uncovered, but level may still be above the fuel. A subcooling value of 0°F indicates saturation conditions. While not entirely conclusive, it's likely that the core is still covered.
Plausible; The examinee may believe that the combination of 0% head level and 0°F subcooling is indicative of core uncovery.

C - WRONG; HJTC above saturation simply means it is uncovered, not the core.
Plausible; CETs (NOT HJCTs) at 700°F is an indication of core uncovery per EAL classification tables. RVLMS at 0% means that level is below the lowest head level probe, but the core may still be covered.

D - WRONG; Pressure and CET values indicate the RCS is at saturation, therefore the core is still covered.
Plausible; This appears to be indication of superheated conditions because temperature is higher than pressure. However, this is the saturation point where pressure and temperature cross.

References

EOP-2541, Appendix 2, R2; RCS P/T Requirements and Steam Tables.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 074 Inadequate Core Cooling

Number EA2.01 **RO** 4.6 **SRO** 4.9 **CFR Link** (CFR 43.5 / 45.13)

Ability to determine or interpret the following as they apply to a Inadequate Core Cooling: Subcooling margin

Question #: 85 | Question ID: 1150008 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The plant has tripped from 100% power due to a small tube rupture on the #1 Steam Generator (SG). Upon completing Standard Post Trip Actions, the crew noted that MFW, Condensate and AFW are UNAVAILABLE, and have transitioned to EOP-2540, Functional Recovery procedures.

Several minutes later, the following conditions exist:

- #1 SG level = 35% and stable.
- #2 SG level = 110" and dropping.
- #1 ADV closed.
- #2 ADV open 50%.

The US has decided to transition to the Once-Through-Cooling (OTC) success path of EOP-2540D.

Which of the following actions are required for the successful implementation of this success path, and what is the possible consequence if those actions are delayed?

☐ **A** Open both PORVs and cooldown the RCS to minimize void formation and ensure single-phase NC flow, or core uncovery could result.

☐ **B** Initiate an RCS cooldown NOT to exceed 80 °F/hr. and open PORVs, or the Reactor Vessel belt line could exceed design parameters.

☐ **C** Open both ADVs to cooldown the RCS at max rate, then open PORVs at the 200 °F subcool line to prevent PTS of the RCS and reactor vessel.

☑ **D** Open both ADVs 100% and open both PORVs, or core damage may occur due to inadequate safety injection flow.

Justification

D - CORRECT; PORVs and ADVs must both be opened to initiate once-through cooling, or the limited PORV flow capacity will result in eventual core uncovery and fuel damage.

A - WRONG: In an event where a LOOP has resulted in the loss of Force Flow, single-phase NC is most desirable. But again, not in this event.
VALID DISTRACTOR: applicant may confuse a desired condition with a realistically achievable condition.

B - WRONG: These are a possible contingency actions for a SGTR, if RCS or SG pressure is holding up injection flow.
VALID DISTRACTOR: applicant may confuse this "legitimate contingency" for the required actions here.

C - WRONG: The concern is from an ESD event and a possible contingency if the stated conditions cannot be controlled.
VALID DISTRACTOR: applicant may confuse this "legitimate contingency" for the required actions here.

References

1. OP-2260, R9C3; EOP-2537, Loss of All Feedwater, Overview/Strategy
2. EOP-2540D, HR-3, Step 1

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** E09 Functional Recovery

Generic K/A Selected

NRC K/A Generic **System** 2.2 Equipment Control

Number 2.2.44 **RO** 4.2 **SRO** 4.4 **CFR Link** (CFR: 41.5 / 43.5 / 45.12)

Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Question #: N/A | Question ID: 5000008 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 1 | ☐ Selected for Exam | Origin: Parent | ☑ Past NRC Exam?

PARENT

The plant has tripped from 100% power due to a loss of all feedwater. Upon completing Standard Post Trip Actions, the crew has noted that MFW, Condensate and AFW are UNAVAILABLE, and has transitioned to the Loss Of All Feedwater EOP.

Several minutes later, SIAS has been manually imitated and the US has decided to implement the Once-Through-Cooling (OTC) success path.

Which one of the following describes additional actions required for successful implementation of OTC, and the possible consequences if those actions are delayed?

☑ **A** Open both ADVs 100% and open both PORVs, or core damage may occur due to inadequate safety injection flow.

☐ **B** Initiate an RCS cooldown NOT to exceed 80 °F/hr. and Open PORVs, or the Reactor Vessel belt line could exceed design parameters.

☐ **C** Open both PORVs and cooldown the RCS to minimize void formation and ensure single-phase NC flow, or core uncovery could result.

☐ **D** Open both ADVs to cooldown the RCS at max rate, then open PORVs at the 200 °F subcool line to prevent PTS of the RCS and reactor vessel.

Justification

CHOICE (A) - YES
PORVs and ADVs must both be opened to initiate once-through cooling, or the limited PORV flow capacity will result in eventual core uncovery and fuel damage.

CHOICE (B) - NO
WRONG: These are a possible contingency actions for a SGTR, if RCS or SG pressure is holding up injection flow.
VALID DISTRACTOR: applicant may confuse this "legitimate contingency" for the required actions here.

CHOICE (C) - NO
WRONG: In an event where a LOOP has resulted in the loss of Force Flow, single-phase NC is most desirable. But again, not in this event.
VALID DISTRACTOR: applicant may confuse a desired condition with a realistically achievable condition.
CHOICE (D) - NO
WRONG: The concern is from an ESD event and a possible contingency if the stated conditions cannot be controlled.
VALID DISTRACTOR: applicant may confuse this "legitimate contingency" for the required actions here.

References

1. HPI-00-C, "High Pressure Safety Injection System" Lesson, Revision 6, (Pg 12 of 49)
2. CVC-00-C, "Chemical and Volume Control System" Lesson, Revision 8, (Pg 37 of 165)
3. EOP-2540D Functional Recovery of Heat Removal Technical Guide, Revision 18, (Pg 122 of 155)

Comments and Question Modification History

Reworded question to NOT give plant indications th

NRC K/A System/E/A **System** E06 Loss of Feedwater

Number EA1.2 **RO** 3.4 **SRO** 4.0 **CFR Link** (CFR: 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the (Loss of Feedwater) Operating behavior characteristics of the facility.

Question #: 86 | Question ID: 1100043 | ☐ RO ☑ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant has just entered MODE 4, shutting down to refuel. Chemistry then calls and notifies the US that the RWST boron concentration is below the Tech. Spec. limit of 1720 ppm.

Which one of the following describes the Tech. Spec. bases for the RWST boron concentration limit and what action is the US required to now take?

☑ **A** The reactor will remain subcritical following mixing of the RWST and RCS water volumes during a small break LOCA, assuming all CEAs inserted except for the most reactive CEA.
Within one hour, raise RWST boron to the required concentration using OP-2304C, Makeup Portion of CVCS, or cool the plant down to MODE 5 within the next 30 hrs.

☐ **B** The reactor will remain subcritical following mixing of the RWST, Shutdown Cooling and RCS water volumes, with all control rods removed during refueling operations (MODE 6).
Prior to performing any core alterations while refueling, raise RWST boron to the required concentration using OP-2304F, CVCS Operation While In Cold Shutdown.

☐ **C** The reactor will remain subcritical following ECCS injection into the RCS during an Excess Steam Demand Event, assuming all CEAs inserted except for the most reactive CEA.
Within 72 hrs, raise RWST boron to the required concentration using OP-2304C, Makeup Portion of CVCS, or cool the plant down to Mode 5 within the next 6 hrs.

☐ **D** The reactor will remain subcritical following ECCS injection into the RCS during an Anticipated Transient Without Scram (ATWS) with a complete loss of the secondary heat sink.
Within 72 hrs, verify the BASTs contain the required boron by volume, using SP-2601A, Borated Water Sources Verification, or cool down to Mode 5 within the next 36 hrs.

Justification

A - CORRECT; Technical Specification, 3/4.5.4 Basis states, "The limits on RWST minimum volume and boron concentration ensure that... 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and RCS water volumes. SBLOCA accident analysis assumes ARI except for the highest worth CEA. LBLOCA assumes ARO.

B - WRONG; Control rods are not removed from the fuel in the vessel in any MODE.
Plausible; The examinee may believe that a sufficiently high enough boron concentration is maintained in the RCS to allow CEAs to be removed from spent fuel assemblies in the vessel during refueling to permit their transfer to the new fuel assemblies.

C - WRONG; The basis for the RWST spec does not include the mitigation of an ESD. Less RWST water will be injected into the RCS following an ESD then would be injected by a LOCA; therefore a LOCA is more limiting
Plausible; The examinee may believe that the positive reactivity added by the cooldown from and ESD must be counteracted by the injection of RWST water.

D - WRONG; The Charging Pumps taking a suction form the Boric Acid storage Tanks are credited for an ATWS or loss of secondary heat removal.
Plausible; The examinee may be confused about the basis for ECCS equipment. The ECCS spec includes Charging Pumps to mitigate an ATWS or loss of secondary heat sink, but does not necessarily require the RWST as the suction source.

References

Tech. Spec. Bases for 3.5.4, RWST

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 004 Chemical and Volume Control System

Number A2.27 **RO** 3.5 **SRO** 4.2 **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: Improper RWST boron concentration

Question #: 87 | Question ID: 1000062 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 2 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

The unit is at 100% power.
The "A" EDG is out for on-line maintenance in accordance with TSAS 3.8.1.1.b.4.
Surveillance is scheduled on the "D" CAR cooling unit.
During performance of the surveillance, the "D" CAR fan trips when a start in slow speed is attempted.
RBCCW flow through the unit is acceptable and it is verified that the unit will run in high speed.

For this set of conditions you must:

☐ **A** Restore the 'D' CAR cooling unit to Operable status within 7 days or be in Hot Shutdown within the next 12 hours.

☐ **B** Initiate action within 1 hour to place the unit in Hot Standby within the next 6 hours and in Hot Shutdown with pressurizer pressure < 1750 psia within the following 6 hours.

☑ **C** Restore the 'A' EDG or the 'D' CAR cooling unit to Operable status within 2 hours or place the unit in Hot Standby with pressurizer pressure < 1750 psia within the following 6 hours.

☐ **D** Restore the 'D' CAR cooling unit or the 'A' EDG to Operable status within 48 hours or be in Hot Shutdown within the next 12 hours.

Justification

C: CORRECT, the facility 1 CAR cooling train must be considered inop IAW provisions of TS 3.0.5 since emergency power for the 'A' & 'C' is OOS. With the "D" CAR Cooler out of service, Tech Spec 3.0.5. applies

A: WRONG; This is the correct action for ONLY an inoperable CAR Fan.
Plausible: Chosen if examinees think only 'D' CAR fan must be considered inop

B: WRONG; Facility 1 CAR Fans are NOT considered inoperable with only its emergency power supply inoperable; therefore; only the "D" CAR Fan is considered inoperable.
Plausible: Chosen if examinees think TS 3.0.3 applies.

D: WRONG; TS 3.0.5 applies and is more limiting than either the CAR Fan TS or the EDG TS.
Plausible: Chosen if examinees think that only TS 3.6.2.1 applies.

References

Tech. Spec. 3.0.5 and 3.6.2.1.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 013 Engineered Safety Features Actuation System (ESFAS)

Generic K/A Selected

NRC K/A Generic **System** 2.2 Equipment Control

Number 2.2.36 **RO** 3.1 **SRO** 4.2 **CFR Link** (CFR: 41.10 / 43.2 / 45.13)

Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Question #: 88 | Question ID: 1100044 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant tripped from 100% power due to a Large Break LOCA, with the following events and conditions:

- RSST is unavailable, both Emergency Diesel Generators (EDG) started and loaded on the LNP.
- "A" EDG was manually tripped when the "A" Service Water Pump would NOT restart on the EDG.
- 24E is tied to 24C, and both are de-energized.
- 24D is energized on the "B" EDG.
- SIAS, CIAS, EBFAS, MSI and CSAS fully actuated for Facility 2.
- ALL other equipment is operating as designed.

The crew has just started implementing EOP-2532, LOCA, when the "D" CAR Fan trips on overload. The RO reports that containment pressure is 24 psig and starting to slowly rise.

Which one of the following statements describes the course of action the US must take?

☐ **A** Immediately transition to EOP-2540F, CTMT Temperature and Pressure Control, and restore CAR Fans to operation.

☐ **B** Immediately transition to EOP-2540F, CTMT Temperature and Pressure Control, and restore CTMT Spray to operation.

☐ **C** Immediately attempt to energize Bus 24E and 24C from Unit 3 using EOP-2541, then restore Facility 1 CAR Fans to operation using EOP-2532, LOCA.

☑ **D** Immediately attempt to energize Bus 24E and 24C from Unit 3 using EOP-2541, then restore Facility 1 CTMT Spray to operation using EOP-2532, LOCA.

Justification

D - CORRECT: The given conditions will result in a loss of all but one CAR Cooler and the "B" Containment Spray Pump for Containment temperature and pressure control. Action must be taken to restore Additional Containment cooling. Restoration of power to Bus 24C will allow the "A" Containment Spray Pump and the Facility 1 CAR Coolers to be placed in service and preserve the Containment Temperature and Pressure Control Safety Function.

A - WRONG: EOP usage guidelines do not allow direct transition to a specific Functional Recovery Safety Function procedure.
Plausible: This action could possibly succeed, if it were allowed.

B - WRONG: EOP usage guidelines do not allow direct transition to a specific Functional Recovery Safety Function procedure.
Plausible: This action would succeed, if it were allowed.

C - WRONG: RBCCW can not be restored on Facility 1 due to CTMT pressure being >20 psig. Therefore the Facility 1 CAR Fans cannot be recovered.
Plausible: This action would work if it were not for the waterhammer concern in the CAR Coolers.

References

EOP-2532, R29C1, Steps 11, 13 & 36

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 026 Containment Spray System (CSS)

Generic K/A Selected

NRC K/A Generic **System** 2.4 Emergency Procedures /Plan

Number 2.4.1 **RO** 4.6 **SRO** 4.8 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of EOP entry conditions and immediate action steps

Question #: 89 | Question ID: 1100045 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

Which of the following describes the basis for the Turbine Battery Technical Specification?

☐ **A** On a loss of Inverter 1 and 2, concurrent with a turbine trip and subsequent ATWS, power from the Turbine Battery will ensure an AFAS occurs and triggers Aux. Feedwater to feed at least one S/G, preventing a loss of the RCS heat sink.

☑ **B** On a loss of a Vital DC Bus concurrent with an Excess Steam Demand Event in containment, power from the Turbine Battery will ensure an MSI actuation can close the Main Feed Reg. Valves, isolating main feed flow to the affected S/G.

☐ **C** On a loss of a Vital DC Bus concurrent with a Large Break LOCA, power from the Turbine Battery will ensure ESAS can actuate the applicable safety injection systems and components to reflood the RCS and maintain the core covered.

☐ **D** On a loss of Inverter 1 and 2 concurrent with a Large Break LOCA, power from the Turbine Battery will ensure ESAS does NOT prematurely actuate an SRAS on the failure of two RWST level instruments, causing a loss of safety injection.

Justification

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

A - WRONG; This is NOT the bases in Tech Specs.
Plausible: Because the Turbine Battery will supply VA-10 and VA-20 through INV-5 and INV-6 in this case, the examinee may believe it to be the basis from Tech specs.

C - WRONG; Only one Vital DC is required to actuate one complete facility of ESAS.
Plausible: The examinee may feel that ESAS will NOT actuate with the loss of a Vital DC Bus due to the loss of two Vital 120 Volt AC Buses; therefore, the Turbine Battery will provide backup power to at least one of the deenergized buses.

D - WRONG; The Facility 1 and Facility 2 RWST level instruments have backup power from Facility 3 and Facility 4; therefore, the Turbine Battery does NOT provide any backup for the RWST level instruments.
Plausible: The examinee may believe that a loss of Inverter 1 and Inverter 2 requires the backup power from the Turbine Battery to keep the RWST level instruments energized.

References

TS Bases for 3.8.2.5 (Pg. B 3/4 8-18)

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 063 DC Electrical Distribution System

Generic K/A Selected

NRC K/A Generic **System** 2.2 Equipment Control

Number 2.2.25 **RO** 3.2 **SRO** 4.2 **CFR Link** (CFR: 41.5 / 41.7 / 43.2)

Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Question #: 90 | Question ID: 1100046 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant is in Mode 1 and the #1 SIT must be sampled per OP-2306O, "Safety Injection Tanks, RCS >1750 psia", following a recent water addition.
The US is about to conduct a brief on the procedure and administrative requirements for the operation of 2-SI-463, SIT Recirc Header Stop Valve.

Which of the following describe requirements for the operation of 2-SI-463?

☑ **A** A PEO with no other tasks must remain at 2-SI-463, in direct communications with the control room, during the entire evolution where the valve may be operated.
The valve is to be opened and closed only by direction from the control room staff.

☐ **B** A PEO with no other tasks must remain at 2-SI-463, in direct communications with the control room, during the entire evolution where the valve may be operated.
The valve is to be opened or closed on direction of the Chemist drawing the sample.

☐ **C** The PEO sent to operate 2-SI-463 must immediately notify the control room when 2-SI-463 has been opened or reclosed, prior to continuing PEO Rounds.
The valve is to be opened and closed only by direction from the control room staff.

☐ **D** The PEO sent to operate 2-SI-463 must immediately notify the control room when 2-SI-463 has been opened or reclosed, prior to the performance of other tasks.
The valve is to be opened or closed on direction of the Chemist drawing the sample.

Justification

A - CORRECT; 2-SI-463 is a CTMT isolation valve and is required to be locked closed per CTMT integrity requirements. Per OP-2306O, whenever the valve is operated in Modes 1 - 4, a Dedicated Operator must be stationed at the valve, in direct communications with the control room, for the entire time the valve is unlocked or open.

B - WRONG; Operation of 2-SI-463 must be directed from the control room, due to the administrative impact of opening it.
Plausible; Chemistry personnel are authorized to manipulate various valves in the performance of their duties, without direction from the control room on specific valve operation. 2-SI-463 is only opened to allow the Chemist to draw a sample of the SIT and would only be opened when the Chemist communicated that they were ready to draw the sample.

C - WRONG; Operation of 2-SI-463 requires a "Dedicated Operator" (no other duties), due to the administrative impact of opening it.
Plausible; There are many valve operations of Tech. Spec. controlled valves, that do <u>not</u> require a "Dedicated Operator" be present.

D - WRONG; Operation of 2-SI-463 requires a "Dedicated Operator" (no other duties), due to the administrative impact of opening it.
Plausible; Any sampling evolution would normally be controlled by the Chemist, who would dictate when the applicable valves need to be manipulated.

References

OP-2306O, R2C5, Pg. 4, Precaution #2 and Pg. 13, Step 4.3

Comments and Question Modification History

02/01/11; Per validation, changed choice 'C' words from **"the performance of other tasks"** to **"continuing PEO Rounds"**. - rlc.

NRC K/A System/E/A **System** 103 Containment System

Generic K/A Selected

NRC K/A Generic **System** 2.1 Conduct of Operations

Number 2.1.23 **RO** 4.3 **SRO** 4.4 **CFR Link** (CFR: 41.10 / 43.5 / 45.2 / 45.6)

Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Question #: **91** | Question ID: 1180625 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

The plant was operating at 100% power, ARO (180 steps withdrawn).
0730 - A Reg. Group 7 CEA drops to the bottom (0 steps withdrawn) and the US enters AOP-2556, CEA Malfunctions.
0820 - Reactor power is stable at the required level for CEA recovery.
0910 - I&C has repaired the problem with the CEDS and reports the dropped CEA is ready for recovery.
0930 - The dropped CEA has been withdrawn to 162 steps.

Which of the following actions is the US required to take and why?

☐ **A** Enter LCO 3.0.3. and withdraw the CEA to at least 170 steps by 1030; otherwise, the unit must be shutdown to MODE 3 using boration only. This prevents exceeding the LHR setpoints caused by the local power peaks during CEA withdrawal.

☐ **B** The CEA must be withdrawn to at least 170 steps by 1020, or power must be reduced to < 20% using boration only.
This is to prevent excessive flux shifts to the upper region of the core from Xenon buildup, which could challenge DNBR.

☐ **C** Immediately stop withdrawing the dropped CEA and commence a plant shutdown to <20% power using boration only.
This will prevent radial flux shifts generating local power peaks that exceed the design specifications of the fuel vendor.

☑ **D** Immediately stop withdrawing the dropped CEA and commence a plant shutdown to Mode 3 using boration only.
CEA recovery efforts cannot continue in order to prevent fuel damage due to xenon buildup and local power peaks.

Justification

D - CORRECT; TSAS 3.1.3.1 action A.1 is applicable due a misaligned CEA (>20 steps)
AOP 2556 step 4.28.k. IF CEA is not realigned to within 10 steps of all other CEAs in its' group, within 2 hours, PLACE plant in HOT STANDBY condition within the next 6 hours. It's been two hours since the CEA was misaligned and has not bee realigned to within 10 steps of its group; therefore a shutdown must begin. The basis for this actions is the distortion of core power distribution.

A - WRONG; TSAS 3.0.3 is NOT applicable because TSAS 3.1.3.1 provides specific actions to take for this condition.
Plausible: The examinee may feel that the CEA is misaligned for longer than the two hours allowed by TSAS 3.1.3.1; therefore, TSAS 3.0.3 must be applicable.

B - WRONG; The two hour time clock from TSAS 3.1.3.1 starts as soon as the CEA is misaligned by more than 20 steps; therefore, the CEA must be withdrawn to at least 170 steps by 0930.
Plausible: The examinee may believe that the time clock begins as soon the Tech Spec power level is reached.

C - WRONG; Commencing a plant shutdown is appropriate; however, the reason (basis) is incorrect. Radial flux shifts may be produced, but the real concern is axial power distribution.
Plausible: The examinee may believe that design specs could be exceeded due to radial flux redistribution as a result of an extended time with a CEA partially inserted.

References

AOP-2556, R16C10, Step 1.2 Discussion section and the Caution preceding step 4.28

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 014 Rod Position Indication System (RPIS)

Number A2.04 **RO** 3.4 **SRO** 3.9 **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 RPIS; and (b) based on those on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Misaligned rod

Question #: N/A | Question ID: 80625 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 3 | ☐ Selected for Exam | Origin: Parent | ☑ Past NRC Exam?

PARENT

The plant was operating at 100% power, steady state, with all CEAs fully withdrawn.

Then, CEA #1 in Reg. Group 7 drops to the bottom of the core (0 steps withdrawn)
The crew subsequently performs all required actions, per the applicable AOP, and is awaiting the I&C "go-ahead" to begin recovery of the dropped CEA.

It has now been one hour and 50 minutes since the CEA dropped, and I&C has just informed the control room that CEA recovery steps may begin.

Which of the following is the required action?

☑ **A** Immediately commence a plant shutdown to MODE 3 by Boration only.

☐ **B** Withdraw the dropped CEA to at least 170 steps within the next 10 minutes.

☐ **C** Within the next 10 minutes, initiate steps to recover the dropped CEA.

☐ **D** Trip the plant and maintain the reactor shut down for a minimum of 2 hours.

Justification

A - Correct; Per the requirements of TSAS 3.1.3.1 action A.1 and AOP 2556 step 4.28.k: IF the misaligned CEA is not realigned to within 10 steps of all other CEAs in its' group within 2 hours, PLACE the plant in HOT STANDBY condition within the next 6 hours. The remaining 10 minutes is NOT enough time for recovery of the CEA.
B - Wrong; Recovering a CEA this quickly is in violation of the AOP guidance, as it has a strong possibility of damaging the fuel. Based on the AOP recovery guidelines, it is mathematically impossible to recover the dropped CEA within the Tech. Spec. time limit.
C - Wrong; Initiating the recovery of the CEA within two hours does NOT meet the requirements of TSAS 3.1.3.1 action A.1 and AOP 2556 step 4.28.k. The CEA must be within 10 steps if its group within two hours.
D - Wrong; A plant trip is NOT required and would be considered an overly aggressive plant shutdown and a non-conservative action.

References

Tech. Spec. 3.1.3.1 CEA Position, Action "A"; Misaligned by >20 Steps.
AOP-2556; Step 4.28, Misaligned CEA Recovery.

Comments and Question Modification History

Eliminated the references per NRC comment. Changed Distractor "C". Reactor Operators do NOT apply Tech Specs.

NRC K/A System/E/A System 2.2 Equipment Control

Generic K/A Selected

NRC K/A Generic System 2.2 Equipment Control

Number 2.2.40 **RO** 3.4 **SRO** 4.7 **CFR Link** (CFR: 41.10 / 43.2 / 43.5 / 45.3)

Ability to apply Technical Specifications for a system.

Question #: 92 | Question ID: 71489 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 4 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

Given the following caution statement from AOP 2583, Loss of All AC Power During Shutdown Conditions: "DC power conservation requires prompt action within 1 hour"

Which one of the following is a basis for this caution?

☑ **A** With one battery supplying both 201A & 201B busses, the battery is sized to supply the total connected loads for one hour without charger support.

☐ **B** After one hour, the higher current flow caused by the lower battery supply voltage, will approach the current limits of the Vital Instrument AC Inverters.

☐ **C** The estimated time to establish containment closure is one hour and DC power is needed to accomplish all of the tasks necessary for this evolution.

☐ **D** The DC switchgear rooms will have no ventilation and if the batteries are discharged for more than one hour, dangerous hydrogen levels could build up.

Justification

A - CORRECT; T/S bases for 3/4.8 Electrical Power Systems. With one battery supplying both 201A & 201B busses, each battery is sized to supply the total connected loads for one hour without charger support.

B - WRONG; High current flows to the Inverters is not a concern at this time.
Plausible; Current will rise as battery voltage drops with depletion, and the current limiting circuits on the Vital Inverters has been defeated.

C - WRONG; The time requirement to establish CTMT closure is based on the most limiting of RCS Time-To-Boil or 2 hours.
Plausible; DC power is necessary for many components to function, and CTMT closure is required to be set in this scenario.

D - WRONG; Dangerous levels of hydrogen gas are not expected to build up at this time.
Plausible; There are no ventilation fans available for the switchgear and battery rooms and batteries are known to give off hydrogen gas.

References

Tech. Spec. Bases 3.8.2.4, Pg 8-17

Comments and Question Modification History

02/01/11; Per validation, change in correct answer 'A', "**each** battery" to "**the** battery". - rlc.

NRC K/A System/E/A **System** 016 Non-nuclear Instrumentation

Generic K/A Selected

NRC K/A Generic **System** 2.4 Emergency Procedures /Plan

Number 2.4.11 **RO** 4.0 **SRO** 4.2 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of abnormal condition procedures.

Question #: 93 | Question ID: 79037 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 2 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

A plant shutdown is in progress with the following conditions:

- Power level is 90% and lowering.
- "A" & "C" SW pumps in service.
- Screens are running in "Fast".
- Injection temperature is 65 °F.

The Crew is utilizing AOP-2560, Storms, High Winds and High Tides, due to a storm out in the Atlantic. Intake structure PEO reports that excessive seaweed fouling of the intake traveling screens with indications of carryover.

Then, the following occurs:

- Traveling screen D/P High alarm annunciates.
- Intake PEO reports the "C" traveling screen motor is running but the screen is no longer rotating.
- The "C" traveling screen D/P is 28" and rising rapidly.
- BOP reports indication that "C" waterbox is fouling.

What actions should the US direct the crew perform?

☐ **A** Enter AOP 2517, Circulating Water Malfunctions, and stop the "C" Circ pump; then cross-tie the "C" and "D" water boxes to maintain condenser vacuum.

☑ **B** Enter AOP 2517, Circulating Water Malfunctions, and stop the "C" Circ pump; then start additional condenser air removal equipment as necessary.

☐ **C** Enter AOP 2565, Loss of Service Water, and shift to the "B" SW pump in service on Facility 2, then enter AOP-2517, Circulating Water Malfunctions, stop the "C" Circ pump; then cross-tie the "C" and "D" water boxes to maintain condenser vacuum.

☐ **D** Enter AOP 2565, Loss of Service Water, and shift to the "B" SW pump in service on Facility 2, then enter AOP-2517, Circulating Water Malfunctions, stop the "C" Circ pump; then start additional condenser air removal equipment as necessary.

Justification

B - CORRECT; Actions in 2517 state "During periods of actual or predicted severe weather, where fouling is a concern, water boxes should ***not*** be cross-connected."

A - WRONG; Cannot X-tie water boxes due to severe weather.
Plausible; This would be the acceptable action to take, by procedure, if severe weather was not a factor.

C - WRONG; The "C" SW pump should not be challenged once the "C" Circ. Pump is secured, and the condenser water boxes cannot be X-tied due to the severe weather.
Plausible; These actions would potentially help if the screen clogged due to a storm surge that was subsiding (past storm).

D - WRONG; With the Circ. Water pump secured, the "C" SW pump should not be starved for a suction source.
Plausible; The "C" SW pump is in the same bay, and behind the same clogging screen, as the "C" Circ. Water pump. It would be logical to swap SW pumps due to the potential loss of suction or clogged screen collapse.

References

AOP-2517, R0C7, Pg. 13, Caution preceding step 5.4

Comments and Question Modification History

02/01/11; per validation, modified stem wording on carryover indication from **"minor carryover is occurring"** to "reported **indications of carryover**". Also added to stem that **"C" waterbox shows indication of fouling.** - rlc

NRC K/A System/E/A System 075 Circulating Water System

Number A2.02 **RO** 2.5 **SRO** 2.7 **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 circulating water system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of circulating water pumps

Question #: **94** | **Question ID:** 83780 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 0 | ☑ Selected for Exam | **Origin:** Bank | ☐ Past NRC Exam?

The plant is currently at 28% power, starting up after a re-fueling outage, when the BOP notices the following:

- #1 FRV Bypass Valve Switch is in AUTO.
- #1 FRV Bypass valve indicates 80% open.
- #2 FRV Bypass Valve Switch is in AUTO CLOSE.
- #2 FRV Bypass valve indicates 0% open.

What is the concern given the indications above?

☐ **A** FRV Bypass valves are required to be closed above 15% Reactor Power due to the "Main Steam Line Break with a failed MSI" accident analysis.

☑ **B** FRV Bypass valves are required to be closed above 25% Reactor Power due to the "Main Steam Line Break Inside CTMT" accident analysis.

☐ **C** FRV Bypass valves are required to be enabled open until 30% Reactor Power due to the instability of auto feed control at low steam demands.

☐ **D** FRV Bypass valves must be operated in synch, either AUTO or AUTO CLOSE, to ensure accuracy of the feedwater flow input to the calorimetric.

Justification

B - CORRECT; OP-2204, Precaution. 3.6 To remain within the main steam line break inside Containment analysis, opening FRV bypass valve(s) is not allowed when greater than 25% power.

A - WRONG; Accident analysis does not assume an MSI failure.
Plausible; The correct answer does involve the impact on the accident analysis of a Steam Line Break. Also, 15% is the approximate power level that the "reactor trip on a turbine trip" is armed.

C - WRONG; The FRV Bypass valves must be closed above 25% power.
Plausible; The feedwater control at low power levels is inherently unstable due to the effects of increased shrink and swell.

D - WRONG; Both FRV Bypass valves must be in AUTO CLOSE above 25% power.
Plausible; The FRV Bypass valves tap into the Main Feed line just upstream of the feedwater flow venturis. If they tapped in downstream (only a couple feet to one side), this effect would be true.

References

OP-2204, R23C5, Precaution 3.6

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.1 Conduct of Operations

Generic K/A Selected

NRC K/A Generic **System** 2.1 Conduct of Operations

Number 2.1.32 **RO** 3.8 **SRO** 4.0 **CFR Link** (CFR: 41.10 / 43.2 / 45.12)

Ability to explain and apply system limits and precautions.

Question #: 95 | Question ID: 1100047 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

What is the bases for the minimum amount of time the reactor must be subcritical, prior to the movement of irradiated fuel?

☐ **A** Allow a sufficient drop in decay heat to ensure the assumed maximum thermal stress on the fuel clad is not exceeded when filling the RCS, with the RWST at minimum temperature.

☐ **B** Allow a sufficient drop in decay heat to ensure the calculations used for the maximum design capacity of the Shut Down Cooling System are valid.

☑ **C** Allow time for the decay of short-lived fission products, thus ensuring the calculated dose consequences of a fuel handling accident are valid.

☐ **D** Allow time for the decay of short-lived fission products, thus ensuring the calculated maximum time to establish emergency containment closure is not exceeded.

Justification

C - CORRECT; Bases for Tech. Spec. 3.9.3, Decay Time: The minimum requirement for reactor subcriticality prior to movement of irradiated fuel ensures that sufficient time has elapsed to allow the decay heat load of the fuel to be within the assumptions of the spent fuel pool heat load analysis. This minimum requirement also ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products so that the calculated radiological dose consequences of the fuel handling accident are bounding.

A - WRONG; The Tech. Spec. for minimum RWST water temperature is based on a potential restart of the reactor when safety injection occurs during a LOCA.
Plausible; Decay heat load will have an effect on the thermal stress seen by the reactor vessel when cold RWST water hits the fuel.

B - WRONG; The capacity of SFP cooling is the concern.
Plausible; The SDC system is designed to supplement SFP cooling when heat loads in the SFP are very high.

D - WRONG; CTMT closure time requirement is not based of the radiation release of short-lived fission products, but long-lived ones.
Plausible; The quantity of radiation released in a fuel handling accident is a concern in the calculated time for CTMT closure.

References

TS Bases 3.9.3, Decay Time

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.1 Conduct of Operations

Generic K/A Selected

NRC K/A Generic **System** 2.1 Conduct of Operations

Number 2.1.41 **RO** 2.8 **SRO** 3.7 **CFR Link** (CFR: 41.2 / 41.10 / 43.6 / 45.13)

Knowledge of the refueling process.

Question #: 96 | Question ID: 3100015 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 1 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

The Plant has tripped from 100% power with the following complications:

- Two (2) CEAs have stuck fully withdrawn.
- VA-10 was lost at the time of the trip.
- The Charging Header has ruptured resulting in a small-break LOCA.
- A Steam Generator Tube Rupture (SGTR) has occurred on the #1 SG.

EOP-2540 has been entered and a natural circulation cooldown was initiated.
RCS temperature was stabilized with Tcold about 485°F and Thot about 505°F.
Procedural actions were then taken to isolate the SGTR in the #1 S/G.

The US is now evaluating various Technical Specification requirements and the actions to continue the plant cooldown.

Which one of the following statements dealing with Technical Specification requirements applies in the existing situation?

☐ **A** A manual plant cooldown can NOT be started until the RCS has soaked for that time required to meet the RCS Cooldown Limits from 100% power temperatures.

☐ **B** The Charging System can NOT be considered OPERABLE until the Alternate Charging Path has been fully established.

☐ **C** ESAS can NOT be considered OPERABLE until ALL Facility #1 ESAS components have been manually actuated or aligned to their accident position.

☑ **D** Shutdown Margin can NOT be considered met until the RCS boron concentration has been raised to account for both of the stuck CEAs.

Justification

D - CORRECT; The shutdown Margin curves take into account the most reactive CEA has stuck out, but ONLY one CEA. If more than one CEA has stuck out on a trip, RCS boron concentration must be raised to account for all CEAs not fully inserted.

A - WRONG; The temperature used for the Cooldown Limit is "reset" once the RCS cooldown from the ruptured SG is stopped. Therefore, a plant cooldown should continue from 420°F, not to exceed the TS limit from that point on.
Plausible; This would be the required action under non-accident conditions.

B - WRONG; The Alternate Charging Path can NOT be used to take credit as an OPERABLE Charging Path.
Plausible; This is an acceptable method to allow the charging pumps to perform their function and inject boric acid in order to re-establishing Shutdown Margin.

C - WRONG; Facility 1 of ESAS is INOPERABLE because of the loss of VA-10. Manual alignment of equipment does not make it operable.
Plausible; This would be a procedure directed action in this scenario to ensure SSCs perform their design function.

References

OP-2208, R13C12; Pg. 21, St. 4.3.6

Comments and Question Modification History

Originally linked to K/A E09-2.1.33 (2.1.33 was moved to 2.2.42)

02/02/11; Per validation, changed 2nd event from an ESD to a SGTR (and modified final RCS conditions accordingly) to make choice 'A' wrong. Also, restructured stem sentences discussing final conditions to improve understanding of actual conditions. Also changed "Facility **One**" to Facility **#1**" in choice 'C'. - rlc.

NRC K/A System/E/A **System** 2.2 Equipment Control

Generic K/A Selected

NRC K/A Generic **System** 2.2 Equipment Control

Number 2.2.42 **RO** 3.9 **SRO** 4.6 **CFR Link** (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)

Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Question #: 97 | Question ID: 5000047 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 1 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

Refueling is in progress on Unit 2. During normal rounds, the Aux Building PEO reports that the red light on the SFP SW Area Radiation Monitor (RM-8139) local module is illuminated.

Which of the following is a possible reason for the reported indication?

☐ **A** Loss of power to the radiation monitor.

☐ **B** Local horn silence switch in the OFF position.

☑ **C** Rad. Monitor module on RC-14 is in ALARM DEFEAT.

☐ **D** Fuel Area Rad. AEAS switch at ESF sensor cab in INHIBIT.

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) - NO
WRONG: Loss of power extinguishes the light.
VALID DISTRACTOR: Plausible that function of red light is to indicate a loss of power.

CHOICE (B) - NO
WRONG: Local horn silence keyswitch disables the audible alarm but leaves the red light lit.
VALID DISTRACTOR: Plausible that function of red light is to inform that audible is defeated.

CHOICE (C) - YES
Local red light illuminates on sensed high radiation condition at a reading exceeding 50mR/hr.

CHOICE (D) - NO
WRONG: Keyswitch at ESF sensor cabinet functions to inhibit the trip and change logic from 2 /4 to 2 out of 3.
VALID DISTRACTOR: Plausible that red light designed to provide local indication of defeated input to ESAS.

References

RMS-00-C, R7C3, "Radiation Monitoring System" Lesson, Pg. 13, Section 1.a

Comments and Question Modification History

02/01/11; Per validation, changed correct answer from "**actual high radiation condition in spent fuel pool area**" to "**Rad. Monitor module on RC-14 is in ALARM DEFEAT**" to increase difficulty level. - rlc

NRC K/A System/E/A **System** 2.3 Radiation Control

Generic K/A Selected

NRC K/A Generic **System** 2.3 Radiation Control

Number 2.3.5 **RO** 2.9 **SRO** 2.9 **CFR Link** (CFR: 41.11 / 41.12 / 43.4 / 45.9)

Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Question #: 98 | Question ID: 2000021 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 2 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

A fuel handling accident has just occurred in the Spent Fuel Pool with gas bubbling up to the surface resulting in a significant rise in area radiation levels.

Which of the following ventilation system alignments must the US have verified to prevent or mitigate the discharge of activity to the environment?

☐ **A** Ensure EBFS fans are aligned to take a suction on the SFP area.
Ensure Fan F-20, Fuel Handling Area Supply Fan, is aligned to supply outside air makeup.

☑ **B** Ensure EBFS fans are aligned to take a suction on the SFP area.
Ensure Fan F-20, Fuel Handling Area Supply Fan, is off with the outside air supply isolated.

☐ **C** Start additional Main Exhaust Fans, discharging to the Main Exhaust System.
Ensure Fan F-20, Fuel Handling Area Supply Fan, is aligned to supply outside air makeup.

☐ **D** Start additional Main Exhaust Fans, discharging to the Main Exhaust System.
Ensure Fan F-20, Fuel Handling Area Supply Fan, is off with the outside air supply isolated.

Justification

B - CORRECT: An AEAS (Auxiliary Exhaust Actuation Signal) will be either manually or automatically initiated. This 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. This should be verified to ensure the release is mitigated.

A - WRONG: F-20 must be secured because it could pressurize the SFP area and cause air leakage directly to the environment.
Plausible; EBFAS is designed to draw a negative pressure in the area, which would be very uncomfortable to personnel in the area. Fan F-20 is designed to supply fresh air to compensate for uncomfortable environmental conditions.

C - WRONG: This would result in a "ground" release of any fission products to the environment.
Plausible; Running F-20 with additional Main Exhaust fans would quickly purge any radioactive gasses from the building.

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.
Plausible; Main Exhaust is the normal ventilation path for this area and starting additional fans would expedite the removal of any radioactive gasses.

References

AOP-2577, R8C4, Fuel Handling Accident, Attachment 1 & 2

Comments and Question Modification History

02/01/11; Per validation, deleted "**minor**" from the stem. - rlc.

NRC K/A System/E/A **System** 2.3 Radiation Control

Generic K/A Selected

NRC K/A Generic **System** 2.3 Radiation Control

Number 2.3.11 **RO** 3.8 **SRO** 4.3 **CFR Link** (CFR: 41.11 / 43.4 / 45.10)

Ability to control radiation releases.

Question #: **99** | **Question ID:** 1160006 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | ☑ Selected for Exam | **Origin:** Bank | ☐ Past NRC Exam?

The plant was tripped from 100% power due to degraded grid voltage. The RSST was lost on the trip, but all other plant systems responded as designed.
The crew has completed EOP-2525, Standard Post Trip Actions, and has transitioned to EOP-2528, Loss of Offsite Power/Loss of Forced Circulation.

Several minutes after transitioning, the US notices the difference between Thot and CET temperatures is about 15°F.

Which of the following describes actions the US must direct to mitigate this condition, in accordance with EOP-2528 and why?

☐ **A** Stop any cooldown and ensure both steam generator saturation temperatures are within 10°F.
Uneven steaming of the steam generators is uncoupling the RCS loops, resulting in uneven thermal stresses on the reactor vessel.

☐ **B** Energize all available Pressurizer heaters and ensure RCS subcooled per highest Thot.
The low RCS flow rate of NC, combined with a lower RCS pressure, is allowing voids to form in the fuel channels and stagnate flow.

☐ **C** Stop any RCS depressurization and ensure the pressurizer is vented by reopening the steam space sample valve.
The momentary loss of power on the trip closed the PZR sample valve, resulting in degraded RCS pressure control due to a hard bubble.

☑ **D** Increase steaming of the SGs and ensure SG pressure is less than saturation pressure for the existing Tcold.
Not enough heat is being removed from the RCS, resulting in a higher than expected RCS temperatures and diminished NC flow.

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: The problem stems from not removing enough heat from the RCS. Stopping the cooldown will make this worse.
Plausible; This is the course of action when the two RCS loops become "uncoupled" due to uneven steam generator heat removal.

B - WRONG: Voids of sufficient size to affect RCS flow should not have formed under the given conditions.
Plausible; Similar conditions as given in the stem would result if voids were impeding RCS flow, and this choice would be correct if the abnormal delta-T is due to void formation.

C - WRONG: There has not been sufficient time for the buildup of non-condensable gasses in the pressurizer to affect pressure control and void formation in the RCS.
Plausible; The PZR steam space sample valve is left open by procedure, to ensure non-condensibles are vented from the PZR steam space. The plant has had problems with a hard bubble due to the buildup of non-condensibles when the valve was unintentionally closed.

References

EOP-2528, Rev. 18,Pg. 8 of 36, Step 11, Contingency b.1

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.4 Emergency Procedure /Plan

Generic K/A Selected

NRC K/A Generic **System** 2.4 Emergency Procedures /Plan

Number 2.4.6 **RO** 3.7 **SRO** 4.7 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)
Knowledge of EOP mitigation strategies.

Question #: 100 | Question ID: 76423 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 1 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

A hostile force gains access to the Protected Area and commits several acts of sabotage inside of the Unit 2 Turbine Building.
- Unit 3 has suffered a loss of communications
- The Unit 2 Shift Manager has just completed classifying the event and intends to initiate a site evacuation.

Which of the following items would require a delay in the site evacuation?

☐ **A** Waterford dispatch can not provide traffic control at this time.

☑ **B** Security has not accounted for all hostile force members.

☐ **C** Accountability of plant personnel is still being performed.

☐ **D** Operational Support Center personnel are still arriving.

Justification

B - Correct; With members of the hostile force still unaccounted for, FAP-08 states that security is still out searching for them. Even though having hostile personnel on site may trigger the natural "flight" response, if personnel are caught moving about, they could easily be mistaken for a member of the hostile force and fired upon.

A - Wrong: FAP08 gives direction for "spare" SERO personnel to manage traffic if the local police are unavailable at the time.
Plausible; The Waterford Dispatch is responsible for providing police officers to direct local traffic for any required site evacuation.

C - Wrong: Personnel accountability can occur after the evacuation. (step 2.1.6).
Plausible; Accountability is accomplish when an evacuation is required. Also, fellow workers are the best source for determining if all personnel have left non-vital spaces.

D - Wrong: Delaying OSC activation does not delay SERO activation and is not a reason to delay a site evacuation.
Plausible; SERO personnel could be delayed in getting to their positions due to the flood of people leaving the sight. This is a known issue that must be considered, but all OSC personnel are not required to be "on-station" for SERO to be considered activated.

References

MP-26-EPI-FAP08, Evacuation and Assembly

Comments and Question Modification History

Question replaced based on previous reviewer comments. 01/06/11
Question modified slightly based on reviews done 01/25/11

02/02/11; Changed event from **"dismissal"** to **"site evacuation"**. Changed all choices to raise difficulty and increase realism.

NRC K/A System/E/A **System** 2.4 Emergency Procedure /Plan

Generic K/A Selected

NRC K/A Generic **System** 2.4 Emergency Procedures /Plan

Number 2.4.40 **RO** 2.7 **SRO** 4.5 **CFR Link** (CFR: 41.10 / 43.5 / 45.11)
Knowledge of the SRO's responsibilities in emergency plan implementation.