MILLSTONE 2011 EXAM –REACTOR OPERATOR WRITTEN EXAM KEY

1.	C	33.	D	65.	A
2.	B	34.	A	66.	D
3.	D	35.	D	67.	B
4.	B	36.	C	68.	C
5.	B	37.	C	69.	B
6.	A	38.	A	70.	A
7.	C	39.	D	71.	A
8.	A	40.	A	72.	D
9.	A	41.	D	73.	D
10.	D	42.	C	74.	B
11.	C	43.	B	75.	C
12.	B	44.	B		
13.	A	45.	A		
14.	B	46.	D		
15.	D	47.	C		
16.	B	48.	A		
17.	D	49.	A		
18.	C	50.	C		
19.	C	51.	B		
20.	A	52.	A		
21.	C	53.	C		
22.	C	54.	D		
23.	B	55.	D		
24.	A	56.	A		
25.	D	57.	B		
26.	B	58.	C		
27.	A	59.	D		
28.	A	60.	C		
29.	B	61.	B – Q#61 Accept B or D		
30.	D	62.	C		
31.	C	63.	D		
32.	B	64.	B		

MILLSTONE 2011 EXAM – SENIOR REACTOR OPERATOR WRITTEN EXAM KEY

1. C
2. B
3. D
4. B
5. B
6. A
7. C
8. A
9. A
10. D
11. C
12. B
13. A
14. B
15. D
16. B
17. D
18. C
19. C
20. A
21. C
22. C
23. B
24. A
25. D
26. B
27. A
28. A
29. B
30. D
31. C
32. B
33. D
34. A
35. D
36. C
37. C
38. A
39. D
40. A
41. D
42. C
43. B
44. B
45. A
46. D
47. C
48. A
49. A
50. C
51. B
52. A
53. C
54. D
55. D
56. A
57. B
58. C
59. D
60. C
61. B – Q#61 Accept B or D
62. C
63. D
64. B
65. A
66. D
67. B
68. C
69. B
70. A
71. A
72. D
73. D
74. B
75. C
76. D
77. B
78. A
79. C
80. B
81. C
82. C
83. A
84. B
85. D
86. A
87. C
88. D
89. B
90. A
91. D
92. D
93. A
94. B
95. C
96. D
97. C
98. C
99. D
100. B

U.S. Nuclear Regulatory Commission

Site-Specific RO Written Examination

Applicant Information

Name: Detailed Answer Key (corrected)

Date: 10/11/2011 (given)

Facility/Unit: Millstone / II

Region: I ☑ II ☐ III ☐ IV ☐

Reactor Type: W ☐ CE ☑ BW ☐ GE ☐

Start Time:

Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination, you must achieve a final grade of at least 80.00 percent. Examination papers will be collected 6 hours after the examination begins.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results

Examination Value	______	Points
Applicant's Score	______	Points
Applicant's Grade	______	Percent

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).

Question Misc. Info: MP2*LORT*5123 [061 AFW-01-C 2530] (1/21/97) 2322, AFAS C99502, NRC-2005 [K/A 061, AFW, A2.05], NRC-2011

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

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

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

of VA-20.

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.

8/29/2011; Per NRC comment in August 2011, Removed space in Choice D.

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. 4 | ☑ 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 open.

Thirty minutes after the trip the following indications are noted:
- RCS pressure is 500 psia and stable.
- Containment/Quench Tank pressure is 30 psig and slowly dropping.
- CET temperatures are 447°F and stable.
- RCS subcooling is 20°F and stable.

Which one of the following PORV discharge temperatures would be indicated if a PORV were stuck partially open?

☐ **A** ~467°F

☑ **B** ~340°F

☐ **C** ~325°F

☐ **D** ~274°F

Question Misc. Info: MP2*LOIT, QT, PZR, RCS, PORV, MB-05424, NRC-2011
Requires use of Steam Tables

Justification

B is correct. The leaking PORV would be an isenthalpic process; therefore, the temperature downstream of the open PORV would be based on the enthalpy of steam at 500 psia, taken to the pressure of CTMT, 45 psia (30 psig + 15 psi convert to absolute). That enthalpy at that pressure would equate to a temperature of about 340°F.

A is incorrect. The pressure the PORV is discharging to must be considered.
Plausible; This is the saturation temperature for 500 psia.

C is incorrect. The pressure downstream of the PORV must be converted to absolute pressure.
Plausible: This temperature would be arrived at if 30 psia was used as a down stream pressure.

D is incorrect. PORV discharge enthalpy must be accounted for, not just the pressure it is discharging to.
Plausible: This is the saturation temperature for 45 psia.

References

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

Comments and Question Modification History

08/01/11; Per NRC comments, removed concept used for calculated temperature in each choice.

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 #: 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 coast down 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.

Question Misc. Info: MP2*LOIT, EOP 2532, 2525, SBLOCA, reflux circulation , MB-05940, NRC-2011

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 Iodine scrubbing, it is NOT a requirement. In fact, it is desirable to maintain 40-45% SG level in the affected S/G for Iodine 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.

Question Misc. Info: MP2 LOIT, EOP 2532, LOCA, MB-04749, NRC-2011

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 #: 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 Low Bleedoff Flow 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.

Question Misc. Info: MP2*LOIT, RCP, Seals, NRC-2011

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 was 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.

07/18/11; Per NRC comments, capitalized proper names consistently. - 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 4% every 10 minutes.

☐ **D** Raised the output, VCT now dropping at 1% every 10 minutes.

Question Misc. Info: MP2 *LOIT, PZR, PPLC, Charging, Letdown, NRC-2005 [K/A 022, 2.2.2], NRC-2011

Justification

A - CORRECT; In order to stabilize PZR level without changing Charging Flow (based on given conditions) controller HIC-110 output must be lowered to reduce letdown flow rate. With NO VCT makeup flow, less water returning to the VCT from letdown flow and a constant loss from charging flow, VCT level must drop. The rate of VCT level decrease will be proportional to the level decrease of the PZR due to the RCS leak. Under the stated plant conditions, the VCT is about 1/2 the volume of the PZR. Therefore, the VCT level will decrease at approximately two times the prior rate of pressurizer level decrease, or 4% every 10 minutes.

B - 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.

C - 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.

D - 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. - rlc
07/18/11; Per NRC comments, modified Justification to better explain how the relationship between Charging Flow and PZR level affects the VCT level decrease. - rlc
09/30/11; per NRC comments, changed percentage rate VCT lowers in choices "C" & "D" to match chioces "A" & "B". - rlc

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 <u>and</u> stop the cooldown to prevent a low temperature, over pressure condition in the SDC System.

☐ **B** Start a second LPSI Pump <u>and</u> 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 <u>or</u> close the SDC Suction Isolation Valves to prevent over pressurizing the SDC system.

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

Question Misc. Info: MP2*LOIT, SDC, 2207, MB-05118, NRC-2011

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 #: 8	Question ID: 1100004	☑ RO	☐ SRO	☐ Student Handout?	☑ Lower Order?
	Rev. 1	☑ 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 slowly.
- RBCCW Surge tank level is rising and lowering on the opening and closing of the auto makeup valve.

Which of the following actions are required in accordance with AOP 2564, Loss of RBCCW?

☑ **A** The "A" CEDM Cooler supply and return valves from the "A" RBCCW Header must be closed, the "A" CEDM fan secured and the "B" & "C" CEDM fans verified in service.

☐ **B** The "A" RBCCW Header supply and return header isolations to Containment must be closed which will require the "A" and "C" CAR Fans to be tripped.

☐ **C** The "A" RBCCW Header supply and return header isolations to Containment must be closed which will require the "A" and "C" RCPs to be tripped.

☐ **D** The 3 CEDM Cooler's supply valve and return valve from the "A" RBCCW Header must be closed and the "B" RBCCW Header supply and return valves to the CEDM Coolers must be opened.

Question Misc. Info: MP2*LOIT, RBCCW, AOP, 2564, NRC-2011

Justification

A - CORRECT; This is indicative of a minor 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 be closed individually to prevent the RCPs from being affected.

B - WRONG; The "A" RBCCW header isolation valves that isolate RBCCW to the CEDM coolers do not isolate RBCCW flow to the "A" & "C" CAR Fans.
Plausible: Examinee may confuse the RBCCW CTMT isolations for the "A" and "C" CAR Fans with the RBCCW Header supply and return isolations to CTMT, as these valves are rarely operated.

C - WRONG; This action is not driven by the AOP as it would require a plant trip for a minor RBCCW leak to a non-vital load.
Plausible; Examinee may believe that the CEDM Coolers can only be isolated from outside of Containment, like the CAR Fans.

D - WRONG; The RBCCW isolation valves that would get all three coolers would isolate RBCCW to other components not directly impacted by the leak.
Plausible; Examinee may confuse the RBCCW system valve arrangement for the CEDM coolers with other non-vital components.

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.

07/22/11; Per NRC comments; The justification was changed to reflect that isolation of the "A" CEDM Cooler using the RBCCW Header Isolation will also isolate RBCCW to the "A" and "C" RCPs. Reworded Choice C to isolate only the "A" CEDM Cooler. This was done to ensure choices C and D are not similar. Eliminated "A" and "C" RCPs from distractor B. Changed the justification to reflect this change.

09/05/2011; Reworded all choices to provide only the actions needed to address isolation of the leak and the direct consequences (i.e., removed the requirement to trip the plant as this should be obvious when/if other actions are taken).

09/27/11; per NRC comments, modified question and choices to ensure correct answer is bounded by applicable AOP.
Deleted extra space in choice "B" - 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. 1	☑ 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 (as if a "high pressure" condition existed). 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 were taken?

☑ **A** Only the pressurizer Backup Heaters would deenergize and RCS pressure would lower causing Proportional Heater output to rise and stabilize RCS pressure at approximately 2200 psia.

☐ **B** Only the pressurizer Backup 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.

☐ **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.

Question Misc. Info: MP2*LOIT 2304A, PLPCS, VR-21, 2504B, NRC-2011

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; The High Pressure bistable overrides the Backup Heater Low Pressure bistable, preventing the Backup Heaters from reenergizing and helping to stabilize pressure.
Plausible; The examinee may confuse which bistable overrides which, and believe the system will respond as it is designed to for a failure of the "in-service" pressure controller, by energizing the Backup Heaters on low pressurizer pressure.

C - Wrong; The bistable/relay triggered only trips the Backup Heaters, NOT the Proportional Heaters.
Plausible; This would be true if the examinee believes this relay trips all heaters and, therefore, would be overridden by the pressure control system.

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 and it cannot be overridden by any signal as the setpoint is only about 45 psi below the RCS High Pressure Trip setpoint, 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

Comments and Question Modification History

07/18/11; per NRC comments, reworded stem and choices "A" and "B" to improve understanding of how the system is design and, therefore, how the question matches the K/A. - rlc

08/29/2011; Per NRC comment in August 2011, corrected typo in stem.

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 #: 10	Question ID: 1140006	☑ RO	☐ SRO	☐ Student Handout?	☑ Lower Order?
	Rev. 1	☑ 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 insertion of the CEAs when the DSS opens the CEDS MG Set supply breakers.

☐ **B** Initially rise due to the lower production of Xenon and higher RCS pressure, then drop quickly due to the insertion of the CEAs when the DSS opens the CEDS MG Set Output Contactors.

☐ **C** Initially lower due to the effects of the moderator and fuel temperature coefficients, then drop quickly due to the insertion of the CEAs when the DSS opens the CEDS MG Set supply breakers.

☑ **D** Initially lower due to the effects of the moderator and fuel temperature coefficients, then drop quickly due to the insertion of the CEAs when the DSS opens the CEDS MG Set Output Contactors.

Question Misc. Info: MP2*LOIT, CEDM, CEA, 2302A, ATWS, NRC-2011

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 trips the MG set output contactors.
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 inserts the CEAs by causing a loss of the MG sets.

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 the CEAs by completely deenergizing the MG sets.
Plausible: The examinee may believe that the DSS trips the MG set supply breaker, which is controlled by a switch just above the CEA control insert on main Control Board C-04.

References

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

Comments and Question Modification History

07/19/11; Per NRC comments, reworded all choices to improve plausibility. - rlc

8/29/2011; Per NRC comment in August 2011, removed "480 VAC" from Choices A and C. - RJA.

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 per EOP 2534, 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.

Question Misc. Info: MP2, LOIT, SGTR, 2534, NRC-2011

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.
07/19/11; Per NRC comments, added **"per EOP 2534"** to the last sentence in the stem. Did not revise question. - rlc

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.

Question Misc. Info: MP2*LOIT, EOP 2525, 2536, ESD, AFW, LOIT-2004, NRC-2011

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 #: 13	Question ID: 1171926	☑ RO ☐ SRO	☐ Student Handout?	☐ Lower Order?
	Rev. 2	☑ 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 offsite power.
- Bus 24C is deenergized due to a failure of the "A" Emergency Diesel Generator.
- Bus 24E is aligned to Bus 24C.
- "B" AFW pump and the Turbine Driven AFW pump have both failed and are unavailable.
- #1 S/G level is at 90 inches and will drop to 70 inches in 8 minutes.
- #2 S/G level is at 120 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 has entered EOP 2537, Loss of All Feedwater, and is presently reviewing Safety Functions with the STA.
- All other plant systems and components are operating or available as designed.

Which of the following is required per the applicable procedures and why?

☑ **A** The US must immediately direct the RO to initiate Once-Through-Cooling before either SG level reaches 70", because the loss of one facility requires it be initiated at this time to ensure long term core heat removal will be maintained.

☐ **B** The US must immediately direct the RO to assist the BOP with the restoration of power before both SG levels drop below 100", because the loss of one facility will prevent Once-Through-Cooling from successfully maintaining long term core heat removal.

☐ **C** If 24C is not restored in 9 minutes, the US must immediately direct the RO to initiate Once-Through-Cooling , because the loss of one facility requires it be initiated at this time to ensure long term core heat removal will be maintained.

☐ **D** To ensure 24C is restored before either SG level reaches 70", the US must immediately direct the RO to assist the BOP with the restoration of power, because the loss of one facility will prevent Once-Through-Cooling from successfully maintaining long term core heat removal.

Question Misc. Info: MP2*LOIT, MB-05961, NRC, 2537, Main Feedwater, NRC-2011

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; Once-through-Cooling must be initiated early to ensure adequate heat removal with only one HPSI available
Plausible; The examinee may believe that because the restoration of power before 70" is reached is the preferred option, this would be the correct course of action.

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 expedite this task.

References

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

Comments and Question Modification History

09/01/11; Per NRC comments, revised question to improve plausibility of choices and make only one answer correct. - rlc

09/19/11; per Exam Validation, modified the directed operator in choices "A" & "C" from "**BOP**" to "**RO**". It was pointed out that in all training environments, the RO is the designated operator to initiate Once-Through-Cooling by opening the PORVs, unless the RO is not in the control room. Also, changed the time limit in choice "C" from "**8** minutes" to "**9** minutes" to ensure understanding of the assumption solicited in the choice, that 23C is not going to be restored before the first SG reaches 70% level. - rlc

Question #: **13** **Question ID: 1171926** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☐ **Lower Order?**

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

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.

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 <u>all</u> systems and components are functioning as designed.
Assume <u>no</u> 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** Drops suddenly, then quickly rises and stabilizes on automatic operation of the Atmospheric Dump Valves.

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

Question Misc. Info: MP2*LOIT, LNP, 2528, RRS, Tave, NRC-2011

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

07/19/11; Per NRC comments, changed **"Drop"** in choices 'C' & 'D' to **"Drops"**. - 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. 1	☑ 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 <u>both</u> Main Feed Regulating Valves.

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

☐ **C** Automatic control of S/G Feed Pump speed and manual control (C05) of <u>both</u> Main Feed Regulating Valves.

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

Question Misc. Info: MP2*LOIT, AOP 2504C, MFW, VIAC, NRC-2011

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 <u>not</u> 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 <u>not</u> the suggested action of the procedure.
Plausible; The examinee may believe this to be the preferred action because auto SGFP speed control would dampen out course valve operation.

C - WRONG; Manually control of #1 MFRV from C05 is not possible because the loss of VA-10 deenergizes the control circuit. Plausible; The examinee may remember the AOP directs manual control of Main Feed components.

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

07/19/11; Per NRC comments, modified choices to improve plausibility. - 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: 1100054	☑ RO	☐ SRO	☐ Student Handout?	☐ Lower Order?
	Rev. 1	☑ Selected for Exam	Origin: New		☐ Past NRC Exam?

The plant has just tripped from 100% power due to a loss of DV-20.
All other plant equipment functioned as designed, based on the power loss, and the crew has entered EOP 2525, Standard Post Trip Actions.

Which of the following describes actions required in EOP 2525, based on the loss of power, and why?

☐ **A** The "A" Condensate pump must NOT be secured, due to the loss of the "B" and "C" Condensate pumps.

☑ **B** The "B" and "D" RCPs must BOTH be secured due to the loss of Reactor Building Closed Cooling Water flow.

☐ **C** The Turbine Driven Auxiliary Feedwater Pump must be started LOCALLY, due to the loss of control power.

☐ **D** The pressurizer heater control must be shifted to "CHANNEL X" and the breakers reclosed, due to the loss of control power.

Question Misc. Info: MP2*LORT*2796 [063 LVD-01-C 972]copy 2345, NRC-2011

Justification

B - Correct; The loss of DC (control power) will also cause a loss of 24B & 24D on the trip, because the RSST-24D breaker and the "B" D/G output breaker cannot close. With no facility 2 power there is no facility 2 RBCCW, so the two RCPs are running without cooling water and should be immediately tripped manually.

A - Wrong; The "A" & "B" Condensate Pumps are both powered from Bus 25A, which still has power.
Plausible; If DV-10 were lost then this would be a required action.

C - Wrong; The TDAFP does NOT have to started locally because on a loss of DV-20, the TDAFP control power is shifted to DV-10 and can then be operated from C05.
Plausible; The TDAFP is normally powered from DV-20. If control power is not shifted then this action would be required.

D - Wrong; VR-21 has not deenergized on the loss of 24D due to the new UPS power supplies.
Plausible; With all AC power to VR-21 lost, this would be a correct action if the new UPS for VR-21 were not recently installed.

References

AOP 2506B, Loss of DV-20 Load List

Comments and Question Modification History

09/02/11; Per NRC comments, revised question per the following:

- Remove plant conditions from the stem.
- Reworded Stem question statement to improve syntax alignment with the choices.
- Slight rewording of all four choices to improve syntax alignment with the K/A.
- [Did not remove ""All other plant equipment functioning as designed" as this is information to tell examinee that there are no other problems and to focus only on the impact to plant equipment due to the loss of DV-20. MP2 has seen numerous power supply voltage fluctuations and losses due to obsolete system design and natural disturbances. These have resulted in complex system responses when these highly unusual voltage spikes caused individual component fuses and circuit breakers to open. It is important that the examinee not consider historical abnormalities in system response when answering this question.] - rlc

09/28/11; per NRC comments, fixed Justification for choice "A" to match previous changes to the question. - rlc

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: 1150024	☑ RO	☐ SRO	☐ Student Handout?	☐ Lower Order?
	Rev. 1	☑ Selected for Exam		Origin: Mod	☐ Past NRC Exam?

Unit 2 is operating at 100% power, steady state, when a leak develops in a small Instrument Air (IA) line in the Turbine Building. The leak causes an excess flow check valve to close, isolating the leak and a small branch of the IA system. The rest of the IA system is at normal pressure and unaffected by the isolation of the leaking branch.

Which one of the following events could result due to this temporary isolation of IA, and what is the applicable procedure for addressing the problem?

☐ **A** Steam Seal Header Pressure is dropping, address with ARP-2590E (D-38), "STEAM SEAL HEADER PRES HI / LO".

☐ **B** IA pressure to the MSIV #1 is lowering, address with ARP-2590D (C-7), "MAIN STEAM ISOL VALVE 1 AIR PRES LO".

☐ **C** "A" Condenser Steam Dump/Turbine Bypass Valve is opening, address with ARP-2590D (D-6), "CONDENSER BYPASS VALVE NOT CLOSED".

☑ **D** Feedwater Heater 1A Normal Level Control Valve closed, address with ARP-2590D (AA-18), "HEATER 1A LEVEL HI".

Question Misc. Info: (MSS-00-C MB-00231), NRC-2005, NRC-2011

Justification

D - CORRECT; Loss of IA to the FWH 1A level control valves will cause the Normal to fail closed and the High level dump to fail open. Both are addressed by the FWH Level High alarm.

A - WRONG; At 100% power seal leakage through the High Pressure Turbine supplies more sealing steam than the system needs. Even if the supply valve fail closed due to loss of IA, the effect would never be seen at this power level because it is already closed. Plausible: At lower power levels this may be true.

B - WRONG: The MSIVs do not get their air from a small line in the Turbine building. Also, if the IA supply to an MSIV were lost, the valve would go closed and EOP 2525 would be the appropriate procedure.
VALID DISTRACTOR: Loss of IA pressure to an MSIV has happened in the past, causing the valve to close.

C - 'WRONG: The Steam Dump valves fail closed on loss of IA.
Plausible; Steam Dumps are required to open by the FSAR on a plant trip to minimize MSSV lifting. Also, ADVs are required to be OPERABLE by Tech. Specs. The redundancy that is designed into the control system to ensure these valves open on a trip would lend one to believe a simple air line failure would not prevent it.

References

ARP 2590D-073, R0 and Text FWH-00-C, R4, Section C.1.c, HP Feedwater Heaters 1A/B "Control and Instruments".

Comments and Question Modification History

08/10/11; Discovered modified version of question #170 (ID# 1150024) inadvertently deleted or lost from database and "Parent" sent in its place. Modified question replaced in exam and linked to applicable K/A. - rlc

09/19/11: per Exam Validation, corrected typo in correct answer; changed Heater 1**B** to Heater 1**A**. - rlc

09/30/11; per NRC comments, in question stem, removed "Maintenance is able to isolate the leak by crushing the small IA line, but they have not yet followed the line to the specific valve operator it supplies." and added "The leak causes an excess flow check valve to close, isolating the leak and a small section of the IA system." - rlc

NRC K/A System/E/A **System** 065 Loss of Instrument Air

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 #: 18 | **Question ID:** 1100010 | ☑ RO ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 1 | ☑ 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 Generator electrical load to these changes?

☐ **A** Generator Megawatts will rise and MVARs will lower.

☐ **B** Generator Megawatts will rise and MVARs will remain constant.

☑ **C** Generator Megawatts will remain constant and MVARs will lower.

☐ **D** Generator Megawatts will remain constant and MVARs will remain constant.

Question Misc. Info: MP2*LOIT, Generator, 2324, VARs, Frequency, Grid, NRC-2011

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 incorrect. See correct answer C for explanation as to why the basis for voltage and reactive load is correct.
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 correct. See distractor A for explanation as to why the basis for voltage and reactive load is incorrect.
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

07/20/11; Per NRC comments, reworded stem and choices to simplify reading and minimize cues to correct answer. - rlc

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 #58, a Group 3 CEA, is being exercised when it suddenly slips 150 steps and is now only 30 steps withdrawn.
Numerous alarms are received on C-04, including "CEA DROPPED NIS" (CA-18).

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 #58 must be selected on the CEAPDS Backup Scanner.

☐ **B** The Pulse Counter for CEA #58 must be reset on the PPC.

☑ **C** The CEA Motion Inhibit for CEA Group 3 must be bypassed.

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

Question Misc. Info: MP2*LOIT, 2302A, 2556, CEDS, CEA, CMI, MB-05817, NRC-2011

Justification

C is correct. CEA #58 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 #58, it must be bypassed for all of Group 3.

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 A, it has no effect on the CEA #58 individual interlocks.
Plausible: The examinee may believe that this is required because the PPC interlock for Group 3 withdrawal would be armed at this time, preventing Group 3 from being withdrawn.

D is incorrect. The Dropped CEA indication on the RPS/NI channels is not one of the CEDS control signals generated in RPS that inputs to the CEA interlocks.
Plausible: The alarm is triggered if any RPS Narrow Range NI channel detects a 1%/second drop in power. Therefore, it is quite likely that this alarm would be seen when CEA #58 slipped 150 steps, due to it being a peripheral CEA that would shadow a Narrow Range NI. 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#58, 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

07/20/11; Per NRC comments, reworded stem, choices and justification to minimize appearance of 2 correct answers. - 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 inserted <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.

Question Misc. Info: MP2*LOIT*2951 CED-01-C, 2302A, CEDS, NRC, APP, NRC-2011

Justification

The CEA Position Display System (CEAPDS) was installed in the late 1980's to **replace** the obsolete and failing Metroscope, and meet all Tech. Spec. required display (reed), alarm (PDIL) and interlock (deviation) functions of the CEDS.

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).

Comments and Question Modification History

07/20/11; Per NRC comments, added description of how CEAPDS replaced the Metrascope. - rlc

10/04/11; Per NRC comments, corrected minor grammatical typo in stem. - rlc

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. 1	☑ Selected for Exam		Origin: New	☐ Past NRC Exam?

The plant has tripped from 100% power due to grid instabilities and the crew is presently carrying out EOP 2528, Loss of Offsite Power.

Then, VR-21 de-energizes due to a failure of the main feeder breaker to the panel.
The US has directed that Charging and Letdown be secured and that pressurizer level be allowed to cycle on the first backup charging pump.

Assuming Channel "Y" is selected as the controlling channel of pressurizer level, which of the following describes why this direction was given?

☐ **A** Letdown isolates on failure (de-energized) of the 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** Letdown flow goes to minimum and all available (backup) Charging Pumps start. Charging must be secured to limit thermal shock to the letdown side of the Non-Regenerative Heat Exchanger.

☑ **C** The Letdown Heat Exchanger Temperature Control Valve fails closed (on loss of power). Charging and Letdown are secured to prevent flashing down stream of the Letdown Heat Exchanger.

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

Question Misc. Info: MP2*LOIT, 2304A, PLPCS, PZR Level, 2504B, NRC-2011

Justification

C is correct. The Letdown Heat Exchanger Outlet Temperature Transmitter fails causing the 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.

D 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.

07/20/11; Per NRC comments, added **"Assuming Channel "Y" is selected as the controlling channel of pressurizer level,"** to the stem. Also, swapped choices "C" and "D" to even out the number of times that each is correct on the RO exam. - rlc

08/01/11; Per NRC comments, modified stem to solicit knowledge of actions for control system failure in EOP space.

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. 3 ☑ 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 must 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, then recommence only after repairs are made to the discharge monitoring equipment.

Question Misc. Info: MP2*LOIT, 2617A, CLRWS, MB-04398, NRC-2001, NRC-2002 [K/A 059 Accidental Liquid RW, AA1.03], NRC-2011

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; Per SP-2617A, the discharge may be restarted provided the flow rate is calculated based on the actual level change.
Plausible: examinees may chose this distractor if they feel that a failed flow monitor is as bad as a failed Rad. Monitor, which would require a new permit or the instrument be repaired to recommence.

References

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

Comments and Question Modification History

07/20/11; Per NRC comments, modified Choice "D" to improve plausibility. - 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.

Question Misc. Info: MP2*LOIT, fire, 2559, MB-05666, NRC-2002 [K/A 067, Plant Fire, AK1.02], NRC-2011

Justification

All operator actions pertaining to a fire on site are contained in either AOP 2559 or the Appendix 'R' procedure set, AOP 2579A-T.

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.

Comments and Question Modification History

09/01/11; Per NRC comments, explained in Justification why questions pertaining to actions for a plant fire must utilize steps in an AOP instead of an EOP.

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 LOCA inside containment.
Containment pressure has peaked above the setpoint for SIAS actuation.

How does the difference in the automatic system response to a LOCA in the existing mode, as compared to Mode 3 or higher, affect the mitigating strategy?

☑ **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.

Question Misc. Info: MP2*LOIT, LBLOCA, 2207, manual ESAS, MB-05326, NRC-2011

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.

07/20/11; Per NRC comments, made slight modification to last sentence of the stem (clarified question delt with "automatic" system response). - 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

Question #: 24	Question ID: 1183154	☑ RO	☐ SRO	☐ Student Handout?	☑ Lower Order?
	Rev. 1	☑ Selected for Exam		Origin: Mod	☐ Past NRC Exam?

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: 1100056	☑ RO	☐ SRO	☐ Student Handout?	☑ Lower Order?
	Rev. 0	☑ Selected for Exam		Origin: New	☐ Past NRC Exam?

Which of the following control room annunciators require the Plant Process Computer (PPC) in order to alert the operator to an abnormal condition?

☐ **A** C-05 (C-6), CONDENSER STEAM DUMP VALVE NOT CLOSED

☐ **B** C-04 (CA-15), RX LEVEL LOW-FAC. 1

☐ **C** C-08 (C-12) VR11 UPS TROUBLE and (D-12) VR21 UPS TROUBLE

☑ **D** C-06/7 (CB-19) N-16 ALERT and (CA-19) N-16 HIGH

Question Misc. Info: MP2*LOIT/LOUT, SRO, 2534, 2536, 2540, SGTR, ESDE, FRP, (CFR-55-43.b.5), MB-05977, NRC-2011

Justification

D - CORRECT; The PPC monitors the N-16 Rad. Monitors and performs the applicable calculations to determine the actual leak rate. Based on the calculated leak rate an Alert or High alarm is triggered. The N-16 detector outputs can be read at RC-14, but unlike other radiation monitors displayed on RC-14, the N-16 Alert and High settings are not based purely on a radiation level due to their direct link to reactor power level.

A - WRONG; The Condenser Steam Dump Valves' (CDVs) Not Closed annunciator is triggered by any dump valve stem releasing its limit switch as the stem moves upward in the open direction.
Plausible; The CDVs position indication can only be seen on a computer display and have no direct indication on any control room panel.

B - WRONG; Rx Vessel level is read exclusively on the PPC by all in the control room, but it can be read at the ICC cabinet in the old computer room if the PPC is lost.
Plausible; This requires the use of the ICC procedure (not normally used at power) and going through several menus. Also, it is almost always performed by the STA, who monitors the level for Safety Function success determination.

C - WRONG; Trouble alarm (annunciator C-12 & D-12) are driven directly by the UPS control circuitry.
Plausible; The status and all alarms for the VR-11/21 UPS units are monitored on the PPC. The Trouble alarm is there to alert the operators to look at the display and find out what is going on. There is also a PPC alarm that will be generated to do the same thing, in case the control board annunciator is missed among many alarms or fails.

References

C-06/7, ARP 2590E-093, R0C3, (CA-19) N-16 High and 2590E-094, R0C2, (CB-19) N-16 Alert, Initiating Device for both is given as the "PPC".

Comments and Question Modification History

07/22/11; Per NRC comments, generated new question to improve clarity of question and K/A match (<u>not</u> part of NRC comments). - rlc

8/29/2011; Per NRC comment in August 2011, corrected typo in stem (computer). Removed hyphen between VR11 and VR21. Removed periods at the end of Choices A, C, and D. Swapped Choices C and D, along with Justifications. - RJA

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 #: 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 event occurs in Containment and the Reactor is tripped. During the performance of EOP 2525, Standard Post Trip Actions, the RO observes the following plant conditions:

- Four (4) CEAs are stuck fully withdrawn.
- Bus 24A & 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 Facility 1.
- "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 must the RO utilize to mitigate the stuck CEAs?

☐ **A** AOP 2558, Emergency Boration.

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

☐ **C** EOP 2541, Appendix 23, Restoring Electrical Power.

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

Question Misc. Info: MP2 LOIT/LOUT, SRO, E25-01-C MB-2532, 10CFR43(b)(5), MB-05433, NRC-2002, NRC-2011

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

02/03/10; Chip Griffin: Correct answer modified to "EOP 2541, Appendix 3".

07/18/11, Per NRC comments, Added procedure in use in the stem (EOP 2525); changed "should" in question to "must"; in correct answer (A), changed "Appendix 3A" to "Appendix 3" and corrected procedure name (Corrected all procedure names); removed space from "24_A"; added "event" to "Excess Steam Demand". RJA

09/19/11; per Exam Validation, corrected minor nomenclature error in 5th bullet of stem. ("B" charging pump aligned to **Facility 1**)

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 (prior to Pressurizer level control system response) after the leak starts?

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

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

☐ **C** Regenerative Heat Exchanger Outlet Temperature (TI-221) lowers.

☐ **D** Letdown Heat Exchanger Outlet Temperature (TI-224) lowers.

Question Misc. Info: MP2*LOIT*3227 [000 568-01-B 1488] (1/11/95) 2568, AOP, APP, NRC-2011

Justification

A - CORRECT; Letdown flow through the Regen HX transfers heat to Charging. If the heat transfer rate remains the same, then initially, a lower Charging flow through the Regenerative Heat Exchanger would result in a higher Charging differential temperature through the HX . The Charging inlet temperature, which comes from the VCT, remains the same; therefore, Charging outlet temperature must be higher. After one minute, lower Charging flow to the Pressurizer will result in level lowering which will cause Letdown flow to lower slightly. Ultimately, the slightly lower Letdown flow will result in less heat transfer to Charging, resulting in Charging temperature eventually lowering somewhat, but will remain at a higher value than prior to the leak. When Pressurizer level stabilizes at some lower value due to the operation of the Pressurizer level control system, Letdown flow and Charging flow will be equal; however, VCT level will continue to lower due to the 10 gpm Charging header leak.
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 less heat transfer from the Letdown System; therefore, higher Letdown temperature.

C - INCORRECT; Regenerative Heat Exchanger Outlet Temperature is higher due to the lower Charging flow through it, initially.
Plausible: The examinee may think that a leak in the Charging line (lower flow through the Regenerative Heat Exchanger) would cause a reduction in heat transfer from Letdown to Charging; therefore, Regenerative Heat Exchanger Outlet Temperature would have to be lower.

D - 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 a lower heat transfer coefficient; therefore, a lower Letdown temperature.

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"**.

07/18/11; Per NRC comment, changed to higher order vs. memory level. Changed distractors C and D to be more symetrical. (Same temperature instruments as A and B, but with a lowering trend.) Added addition information to correct answer justification.

8/29/2011; Per NRC comment in August 2011, revised 7/18/2011 comment to "changed to higher order vs. memory level" and corrected typo (instruments) in previous comment. Removed extra line between Choices C and D. - RJA

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: 1100053	☑ RO	☐ SRO	☐ Student Handout?	☑ Lower Order?
	Rev. 1	☑ 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.

Assuming all other systems function as designed, which of the following describes parameter response <u>within the first minute</u> after the loss of the 6.9 kV buses?

- ☑ **A** The difference between Th and Tc will be lowering; S/G pressure will be stable or rising slightly.
- ☐ **B** The difference between Th and Tc will be rising; S/G pressure will be stable or rising slightly.
- ☐ **C** The difference between Th and Tc will be lowering; S/G pressure will continue to lower.
- ☐ **D** The difference between Th and Tc will be rising; S/G pressure will continue to lower.

Question Misc. Info: MP2*LOIT, RCS, RCP, RPS, NRC-2011

Justification

A - CORRECT; The response of Th and Tc is due to the design coast down of the RCPs which lasts approximately 1-1.5 minutes. Although both temperatures will be lower, Th will lower faster than Tc due to the sudden, significant reduction in heat generated by the reactor. Tc will stop lowering when the quick open signal is removed (within one minute). S/G pressure will be relatively stable. The Atmospheric Dumps will lower S/G pressure initially, but will quickly stabilize or may rise slightly until stable after the quick open signal is removed and the atmospheric dumps modulate to control pressure.

B - WRONG; Th and TC will initially rise on the loss of RCS flow, but when the Reactor trips, RCS temperatures will lower due to the loss of heat input. Delta-T will NOT be higher than 100% power operation.
Plausible; If the examinee believes that when RCS flow stops, Th will rise or remain the same while Tc lowers in response to opening of the steam dumps.

C - WRONG; Delta-T will lower; however, S/G pressure will NOT continue to lower.
Plausible; The examinee may believe that the opening of the steam dumps and/or safeties will cause S/G pressure to continue to lower.

D - WRONG; Th and TC will initially rise on the loss of RCS flow, but when the Reactor trips, RCS temperatures will lower due to the loss of heat input. Delta-T will NOT be higher than 100% power operation.
Plausible; If the examinee believes that when RCS flow stops, Th will rise or remain the same while Tc lowers in response to opening of the steam dumps.

References

1. RPS-01-C, R6, Pg. 8, b. Setpoint Bases 3) and Pg. 17, b. Design and Operating Characteristics, 1) RCP Underspeed.
2. E28-01-C EOP 2528 PowerPoint, Slide #29.

Comments and Question Modification History

8/29/2011, Revised question, answers and justification per NRC Comments on August 2011. - RJA

09/19/11; per Exam Validation, modified stem question startement from "response **<u>one minute</u>** after the loss" to "response **<u>within the first minute</u>** after the loss" to improve technical accuracy. - rlc

09/28/11; per NRC comments, deleted extra space in choices "B" & "D". - rlc

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?

With the plant operating at 100% power and all systems aligned normally, numerous alarms for Reactor Coolant Pump (RCP) "A" are 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:

RCS Pressure	2250 psia
MIDDLE SEAL	720 psig
UPPER SEAL	165 psig
VAPOR SEAL	10 psig

Based on this data, how many RCP seals are considered either failed or just degraded, per OP 2301C, Reactor Coolant Pump Operation?

- ☐ **A** Seals Failed = 2
  Seals Degraded = 0
- ☑ **B** Seals Failed = 2
  Seals Degraded = 1
- ☐ **C** Seals Failed = 1
  Seals Degraded = 0
- ☐ **D** Seals Failed = 1
  Seals Degraded = 1

Question Misc. Info: MP2*LORT* 2301, RCS, RCP, NRC-2011

Justification

The following D/P's exist: Lower = 1515 psid, Middle = 555 psid, Upper = 155 psid
B - CORRECT; RCP seal alarms, sump level going up, rad monitor alarms, and low Vapor Seal pressure are indicative of a Vapor Seal failure. The failure is NOT catastrophic, but still a failure. The Upper Seal has a D/P of less than 200 psid and is, therefore, considered failed. The Middle Seal has a delta-P greater than 550 psid and is; therefore, OK. Additionally, a lower seal D/P of >1500 psid with one failed seal, indicates the lower seal is degraded.

A - WRONG; The Middle Seal has a delta-P greater than 550 psid and is; therefore, OK. Additionally, a lower seal D/P of >1500 psid with one failed seal, indicates this seal is degraded.
Plausible; Examinee may recognize the two failed seals, but NOT consider the degradation of the Lower Seal since it is still providing a pressure breakdown.

C - WRONG; The Middle seal is OK with the given D/P. The Vapor Seal and the Upper Seal are failed.
Plausible; Examinee may NOT consider the Vapor Seal to be failed because it still is maintaining a 10 psid D/P.

D - WRONG; The Vapor Seal and the Upper Seal are failed.
Plausible; Examinee may NOT recognize both seals as failed. The examinee may consider one of the two failed seals to be functional because it still has some pressure breakdown.

References

OP 2301C, 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.
07/20/11; Per NRC comments: Stem sentence no longer fragmented. Deleted seal pressures at 0000 and 0800. Added OP 2301C, Reactor Coolant Pump Operation. Fixed justifications to match choices. Changed Choice C "=1, =0" (from "=1. =2"). - RJA
8/29/2011; Per NRC comment in August 2011, changed period to comma in stem. Removed reference to time of 166 is stem and in Justification. - RJA
09/29/11; per Exam Validation, corrected math error in stem and Justification. - 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 #: 30	Question ID: 1100016	☑ RO	☐ SRO	☐ Student Handout?	☑ Lower Order?
	Rev. 1	☑ 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.

Which of the following will occur, assuming NO operator action, and what action(s) is (are) required for annunciator C-7 on C-02, VCT Level Lo Lo, per the associated Annunciator Response Procedure?

☐ **A** VCT Outlet Isolation, CH-501, closes and causes Charging Pumps to trip on low suction pressure. Isolate Letdown flow and verify local VCT level matches level indication on C-02.

☐ **B** Makeup Valve Stop, CH-512, opens and initiates a blended makeup directly to the VCT. Monitor Reactor power for any change while the automatic blended makeup is in progress.

☐ **C** Boric Acid Isolation, CH-514, opens and aligns Charging suction to the Boric Acid Storage Tanks. Stop Charging and Letdown and monitor reactor power for changes due to boration.

☑ **D** RWST Isolation, CH-192 opens, and initiates boration from the Refueling Water Storage Tank. Secure Charging and Letdown and adjust Turbine load to stabilize RCS temperature.

Question Misc. Info: MP2*LOIT*1925 [004 CVC-01-C 895] (8/15/96) 2304, CVCS, NRC-2011

Justification

D is correct; at 8% decreasing, LC-227 sends a signal to close the VCT outlet valve, 2-CH-501, and opens the Charging Pump suction flow path to the RWST; 2-CH-504, RWST to Charging Suction (normally open), and 2-CH-192, RWST Isolation.

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 or that the Charging Pump suction flow path is isolated when CH-501 closes.

B is wrong; CH-512 will NOT automatically open on a VCT lo-lo level.
Plausible: With the Makeup Controls aligned for automatic makeup, CH-512 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, Charging Pump suction from the Boric Acid Storage 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".

Comments and Question Modification History

8/29/2011; Per NRC comment in August 2011, changed all choices. - RJA

09/28/11; per NRC comments, removed extra space in choice "A". Verified labeling of CH-512 was correct (Makeup Valve Stop) per OP 2304C and removed the "2-" in front of "CH-512" in choice "B" to match the syntax of the other choices. - rlc

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: 1100055 | ☑ RO ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 2 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant is shutdown for a refueling outage and is presently operating in MODE 5 with Shutdown Cooling (SDC) in service.
Then, the output of FIC-306 fails from 50% to 0%, and SI-306 responds accordingly. All operator attempts to regain control of SI-306 from C-01 are unsuccessful.

Which of the following describes the plant response to this failure and the action required to stabilize RCS temperature, per AOP 2572, Loss of Shutdown Cooling?

☐ **A** Flow through the SDC Heat Exchanger rises and RCS temperature lowers.
Turn the local operator for SI-306 COUNTER CLOCKWISE to stabilize RCS temperature.

☐ **B** Flow through the SDC Heat Exchanger rises and RCS temperature lowers.
Turn the local operator for SI-306 CLOCKWISE to stabilize RCS temperature.

☑ **C** Flow through the SDC Heat Exchanger lowers and RCS temperature rises.
Turn the local operator for SI-306 COUNTER CLOCKWISE to stabilize RCS temperature.

☐ **D** Flow through the SDC Heat Exchanger lowers and RCS temperature rises.
Turn the local operator for SI-306 CLOCKWISE to stabilize RCS temperature.

Question Misc. Info: MP2*LOIT*2912, 2310, SDC, 2572, SI-306, NRC-2011

Justification

C is correct. Shutdown Cooling Total Flow Control Valve, SI-306, is reverse operating on both the controller and the positioner. When controller output goes to 0%, SI-306 will go full open. Because this valve controls total SDC flow by bypassing the SDC heat exchangers, when SI-306 goes open it will result in a decrease in flow through the Heat Exchangers causing RCS temperature to rise. AOP 2572, Loss of SDC, states that, if ONLY SI-306 has failed (due to lost air or power), then RCS temperature must be controlled with SI-657. However, opening SI-657 further will not be sufficient if SI-306 is full open. Therefore, additional temperature control is required, and a PEO must be dispatched to manually close SI-306 locally by turning the operator counter-clockwise to close.

A is incorrect. Flow through the Heat Exchanger will lower and RCS temperature will rise
Plausible: If the examinee believes that, because SI-306 bypasses the Heat Exchanger, it would be logical for it to go closed with a 0% signal, which would result in SDC Heat Exchanger flow rising RCS temperature lowering.

B is incorrect. Flow through the Heat Exchanger will lower and RCS temperature will rise.
Plausible: If the examinee believes that SI-306 is reverse operating on the local operator only. Therefore, it will go closed with a 0% signal and would require the operator be turned in a clockwise direction to reopen.

D is incorrect. Both the controller and the local operator for SI-306 are reverse operating.
Plausible: If the local operator for SI-306 was not reverse operating, this would be correct.

References

1. AOP 2572, Loss of Shutdown Cooling, Step 8.3

Comments and Question Modification History

09/02/11; Per NRC comments, revised question to meet the intent of the NRC suggested wording. Rewording the question exactly as suggested would have been technically incorrect. - rlc

09/28/11; per NRC comments, reworded Justification for choice "B" to clarify and correct the reason for it being incorrect. - rlc

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

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 #: 32	Question ID: 1100059	☑ RO	☐ SRO	☐ Student Handout?	☐ Lower Order?
	Rev. 0	☑ Selected for Exam		Origin: New	☐ Past NRC Exam?

The plant has experienced a reactor trip from 100% power due to a large break LOCA. The following conditions exist one hour after the event:

- All safety related equipment is operating as expected.
- RWST level is 20% and lowering.
- Vital Instrument Panel, VA-10, has just been lost.

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 Isolation, 2-RB-8.1A, must be manually closed.

☑ **B** The "A" LPSI pump must be manually secured from C-01 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 closed by isolating air.

☐ **D** RBCCW to Spent Fuel Pool Cooling Isolation, 2-RB-8.1A must be manually closed locally and SI-659, Minimum Flow Isolation, must be closed by isolating air.

Question Misc. Info: MP2 LORT 2532, NRC-2011

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 loss of both SRAS signals.

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

07/21/11; Per NRC comments; Changed the loss of Actuation Cabinet back to a loss of VA-10 vs. a blown fuse. Removed the word "locally' from chioces A/C/D and improved the method of closing SI-659. Made it clear that VA-10 was lost one hour after the event and before SRAS. Made noun name for RB-8.1A consistent in A and D. "C-01" (NOT C01) is now consistent. - RJA

8/29/2011; Per NRC comment in August 2011, deleted extra space in choice D. Capitalized Minimum Flow Isolation in Choice D. - RJA

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 #: 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 higher and the unidentified leak rate will be the same.

Question Misc. Info: LOIT, 2301A, Quench Tank, 2602A, Leak rate, NRC-2011

Justification

D is correct. 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 a "negative addition" to the system (higher end level subtracted from lower beginning level) and calculates this as a rise in the 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 Unidentified 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

Comments and Question Modification History

09/19/11; per Exam Validation, corrected technical error in correct answer, "identified leak rate will be **higher**", not **lower**. - rlc

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. 4 | ☑ 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~~ two components at the completion of AOP 2564?

1. The position of hand switch "SIAS/LNP ACTUATION SIGNAL HS-6119A", on "B" RBCCW Pump breaker, A504?
2. The "RBCCW PUMP B SIAS/LNP START MANUALLY BLOCKED" annunciator?

☑ **A** 1. BLOCK
2. In alarm

Correction posted for examinees during exam. [illegible] 11/4/11

☐ **B** 1. NORMAL
2. In alarm

☐ **C** 1. NORMAL
2. NOT in alarm

☐ **D** 1. BLOCK
2. NOT in alarm

Question Misc. Info: MP2*LORT,2564, RBCCW, NRC-2011

Justification

A - CORRECT; The final position of the "SIAS/LNP ACTUATION SIGNAL HS-6119A", is dependent on which Facility is supplying Bus 24E, Facility 1 (Bus 24C) or Facility 2 (Bus 24D). In this case, Bus 24C is supplying Bus 24E; therefore, the SIAS/LNP hand switch must be left in the Block position. If Bus 24D is supplying Bus 24E, then the SIAS/LNP hand switch would be placed in the Normal position. Knowing the power supply is key to determining the switch position because the final status of the switch, the annunciator, and the "B" RBCCW Pump on a subsequent SIAS or LNP, is determined by knowing which power supply will allow what configuration. 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 cause an alarm if the pump is running.
Plausible; The SIAS/LNP hand switch is normally in the Block position with NO annunciator. It would be logical to assume that the alarm would NOT be annunciated unless the switch were repositioned.

References

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

Comments and Question Modification History

07/22/11, Per NRC comments; Provided justification as to why the question is a K/A match (test of power supplies). Changed distractors C and D due to implausible distractor D. Also changed justifications for C and D. - rlc

10/04/11, Per NRC comments;

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. 1	☑ 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** "A" RCP Seal Cooler

☑ **D** Blowdown QT Heat Exchanger

Question Misc. Info: MP2*LOIT*3210 RBCCW, 2330A, NRC-2011

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 reactor coolant flowing through the "A" RCP Seal Cooler would be at a higher pressure than the RBCCW system cooling the seal flow. Therefore, any leak that developed would result in flow from the RCS to the RBCCW system.
Plausible: The examinee may equate the low pressure of seal bleedoff in this mode with the actual pressure going through the seal cooler.

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"

Comments and Question Modification History

09/19/11; per Exam Validation, changed the component in choice "C" from "**Primary Sample Cooler**" to "**'A' RCP Seal Cooler**", due to the original component being a possible correct answer in this mode. - rlc

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 #: 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** The spray valves have a small hole drilled into each of their disks
- ☐ **B** A bypass line with an orifice is installed around each spray valve.
- ☑ **C** A bypass line with a valve is installed around each spray valve.
- ☐ **D** The spray valves have a mechanical stop to prevent full closure.

Question Misc. Info: MP2*LOIT, PZR, PPLC, Spray, NRC-2011

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 bypass lines do NOT have an orifice installed..
Plausible: Some systems use an orifice to maintain a set flow or to limit flow.

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

Comments and Question Modification History

07/22/11; Per NRC comments, reworded choices to improve balance.
8/29/2011; Per NRC comment in August 2011, removed extra space in Choice A. - RJA

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.

Question Misc. Info: MP2*LOIT, RPS, CPC, NRC-2011

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** A Main Steam Isolation Signal actuates only on Facility 1.

☐ **B** Breaker A304, Bus 24A to Bus24C Tie Breaker, spuriously trips.

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

☐ **D** Level Safety Channels LT-1113A (#1 SG) and LT-1123A (#2 SG) fail low.

Question Misc. Info: NRC-2005, NRC-2011

Justification

A - 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.

B - 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.

C - 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.

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 each 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, not the same channel on each S/G.

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"

Comments and Question Modification History

07/22/11; Per NRC comments: Changed distractor D to fail both S/G Safety Channel "A" level transmitters.

09/02/11; per NRC, minor rewording of choices to improve LOD. - rlc

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 #: 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 over the next 5 minutes and why?

☐ **A** Amps will decrease due to the rise in Containment air temperature which lowers air density.

☐ **B** Amps will decrease due to the realignment of the Containment Ventilation System performed in EOP 2532.

☐ **C** Amps will increase due to the increase in air flow from opening the spring-loaded discharge dampers.

☑ **D** Amps will increase due to higher air density caused by the increase in the moisture content.

Question Misc. Info: MP2*LOIT*3038 2313, CAR, CTMT, LOCA, NRC-2011

Justification

D is correct. A SIAS will cause the CAR Fans to shift to slow speed while the crew is in EOP 2525. 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: The examinee may feel that the warmer air, being less dense, will result in lower amps.

B is incorrect. Containment ventilation will NOT change in EOP 2532. Performance of EOP 2525 results in a change in Containment Ventilation.
Plausible: The examinee may feel that the additional air flow provided to the CAR Fans will result in lower amps OR that the additional ventilation will result in less restriction in air flow causing amps to lower.

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.

07/25/11; Per NRC comments: Changed question to read, What is the expected response of CAR Fan motor current **over the next 5 minutes** and why? Changed Choice B to read, Amps will decrease due to the **realignment of the Containment Ventilation System performed in EOP 2532**. Removed the words "initially" from Choices A and D. RJA

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 #: 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 EOP 2532, Loss Of Coolant Accident?

☑ **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 SLOW.
Close the "D" CAR Fan Normal RBCCW discharge valve.

☐ **D** Verify the "B" CAR fan is running and start the "D" CAR fan in FAST.
Close the "D" CAR Fan Normal RBCCW discharge valve.

Question Misc. Info: MP2*LOIT, CAR, CCS, 2313A, MB-05425, NRC-2011

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 will become overloaded and the duct work may not be able to withstand the pressure from FAST speed operation.
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 during at power operation).

D - WRONG; Both CAR fans are required to be running in SLOW 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 must be running in FAST to make up for the loss of cooling from the other CAR Fans that are NOT running. Additionally, the examinee may believe that closing the Normal RBCCW Discharge valve will allow more RBCCW flow through the Emergency RBCCW Discharge Valve.

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.
07/25/11. Per NRC comments; Changed Choice C to have D CAR Fan start in SLOW ppeed (vs. FAST). Made first sentence in Choice D the same as Choice B. - RJA
09/19/11; per Exam Validation, changed stem question statement from "per **the applicable EOP**?" to "per **EOP 2532, Loss Of Coolant Accident**?", to better meet question syntax guidelines.

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 #: 41	Question ID: 1100020	☑ RO	☐ SRO	☐ Student Handout?	☑ Lower Order?
	Rev. 0	☑ Selected for Exam		Origin: New	☐ Past NRC Exam?

A Large Break 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, cooling is sufficient to ensure Containment temperature and pressure will remain within 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, cooling is NOT sufficient to ensure Containment temperature and pressure will remain within design limits.

Question Misc. Info: MP2*LOIT, CS, CTMT Spray, 2532, 2309, NRC-2011

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 remain within 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 remain within 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
07/25/11; Per NRC comments: Removed the word "may" from Choice D. Reworded A to "will remain within design limits." Reworded Choice D to , " will remain within design limits." Capitilized the word "cooler" in Choice D to be consistent witah Choice A. - RJA
09/16/11; per Exam Validation, typo in stem, "Large **Beak** LOCA" changed to "Large **Break** LOCA". - 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.
- Vital Instrument Panel, 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, Standard Post Trip Actions, 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 and open both MSIV Bypass Valves, then open the Condenser Steam Dump valves using TIC-4165 on C-05.

☐ **B** Due to the loss of control power to PIC-4216 and the ADVs, utilize the Foxboro Steam Dump Control screen on a PPC work station 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 C-05, 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 on a PPC work station to place the "A" and "B" ADVs in manual and open both ADVs.

Question Misc. Info: MP2*LOIT, LOCA, 2532, Steam Path, NRC-2011

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 used to control one of the steam dump valves.

D - WRONG; "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. - rlc
07/25/11; per NRC comments, reworded choice 'D' to clearly state the action taken would be to open the ADVs. - rja

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.

Question Misc. Info: MP2*LOIT, NRC-2011

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 #: 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 Valve 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 NOT sense the rise in steam flow. The rise in actual steam flow will result in #2 Main Feed Regulating Valve going further open. Level will stabilize at the program setpoint.

☑ **B** The steam flow detectors will NOT sense the rise in steam flow. The level mismatch will generate a signal to open #2 Main Feed Regulating Valve. Level will stabilize below the program setpoint.

☐ **C** The steam flow detectors will sense the rise in steam flow. The resulting rise in Main Feed Pump speed will cause #2 Main Feed Regulating Valve to go further closed. Level will stabilize below the program setpoint.

☐ **D** The steam flow detectors will sense the rise in steam flow. The steam/feed mismatch signal will generate a signal to open #2 Main Feed Regulating Valve. Level will stabilize at the program setpoint.

Question Misc. Info: MP2*LOIT, FRV, MFW, 2321, ADV, NRC-2011

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, with NO rise in feed 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: Even if the examinee does realize the steam flow detectors are downstream of the ADVs then, he/she may think the system will respond to a change in S/G level to maintain S/G levels at setpoint.

C is incorrect. As actual 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: The examinee may remember that feed pump speed rises with a rise in steam flow, but does NOT understand that indicated steam flow will NOT change.

D is incorrect. The steam flow detectors will NOT see the increase in steam flow. As a result, there is NO steam/feed flow mismatch.
Plausible: The examinee may believe that indicated steam flow will rise when the ADV fails open. If this were true, then a steam/feed mismatch would cause the #2 FRV to open and stabilize level at setpoint.

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.

07/25/11; Per NRC comments: Deleted the word "significant" from Choice A; deleted the word "sudden" from Choice D. Changed all Choices as recommended for balance (two have "will sense", two have "will NOT sense".) Changed "valves" in stem to "valve". - RJA

09/01/11; Per NRC comments, modified Choic 'C', 2nd sentence to state the valve would go further closed, not open. Also cleaned up extra carriage returns. - 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 rnust 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.

Question Misc. Info: MP2*LORT*6608 [039 MSS-01-C 928] (10/1/97) 2388, IHES, 2316A, EOP, NRC-2011

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

01/06/11; Modified stem question from "accomplished to close the valves" to "accomplished to allow the valves to be closed", per feedback from Sandy Doboe.

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 #: 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** ONLY the Facility 2 AFW Actuation timer is running and will result in automatic system actuation unless conditions change.

☑ **D** BOTH Facility 1 and Facility 2 AFW Actuation timers are running and will result in automatic system actuation unless conditions change.

Question Misc. Info: AFW-01-C LOIT Auxiliary Feedwater System, NRC-2011

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.

07/25/11; Per NRC comments, Changed Choices A and B to capitalize ONLY and BOTH. Changed Choices C and D to be more balanced.

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 #: 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 step(s) required prior to close A305, 24C/24E Tie Breaker, to energize Bus 24C, assuming no fault on the Bus?

☐ **A** Per EOP 2541, Appendix 23, "Restoring Electrical Power", place all four UV BUS A3 keys in INHIBIT and reset the ESAS UV signal.

☐ **B** Per EOP 2541, Appendix 23, "Restoring Electrical Power", reset the Sequencer on Actuation Cabinet 5.

☑ **C** Per AOP 2502C, "Loss of Vital 4.16 kV Bus 24C", place all four UV BUS A3 keys in INHIBIT and reset the ESAS UV signal.

☐ **D** Per AOP 2502C, "Loss of Vital 4.16 kV Bus 24C", reset the Sequencer on Actuation Cabinet 5.

Question Misc. Info: MP2*LOIT*1971, 2342, IHES, NRC-2011

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 (Bus A3) must be reset prior to energizing Bus 24C. The AOP for loss of Bus 24C would be chosen due to the present MODE of operation.

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. Resetting the Sequencer on Actuation Cabinet 5 is NOT adequate to allow energizing Bus 24C from Bus 24E.
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.
07/25/11: Per NRC comments: Selected as Higher Order. Deleted "reset the ESAS UV signal" from Choices B and D. Reworded second part of stem to, "...the correct step(s)required prior to close A305, 24C/24E Tie Breaker, to energize Bus 24C, assuming on fault on the Bus?" Deleted "...close A305, 24C/24E Tie Breaker." in each of the Choices. Changed "inhibit" in Choices A and C to INHIBIT. - RJA
09/02/11; per NRC, corrected minor typos. - 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 #: 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 loss of pressure in the starting air tanks.

☐ **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 loss of pressure in the starting air tanks.

☐ **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.

Question Misc. Info: MP*LOIT, 125 VDC, 2505, DG, Vital DC, NRC-2011

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". - rlc
02/02/11; **EOP**-2505A should be **AOP**-2505A. - rlc
07/25/11; Per NRC comments, reworded choices "A" and "C" to minimize cues to correct answer. - 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** Kilowatt 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** Kilowatt load will rise; Diesel speed will rise.

Question Misc. Info: MP2*LOIT, EDG, 2346A, NRC-2011

Justification

A is correct. After the output breaker is closed, raising the Governor Control switch will cause kilowatt 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. Kilowatt 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.

09/27/11; per Exam Validation, modified first word in choices 'A' & 'D' from "**Mega**watt" to "**Kilo**watt" to make technically correct. - 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

Question Misc. Info: MP2*LOIT, RM, SGTR, 2383A, NRC-2011

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, 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. RJA

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.

Question Misc. Info: MP2*LORT 2565, SW, Service Water; NRC-2011

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. 1 | ☑ 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). 24E is aligned to 24D with all equipment 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.

A few minutes later, the plant trips due to a loss of the grid (state wide blackout) and all plant systems and components respond as designed.

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.
Both Facilities of SW have flow.

☐ **D** Only Facility 1 RBCCW header has flow.
Only Facility 2 SW header has flow.

Question Misc. Info: MP2*LOIT AOP, 2564, RBCCW, SW, NRC-2011

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 <u>not</u> 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 RB due to starting only "C" RB pump on the "B" EDG.
Plausible; If examinee thinks "C" RB 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

07/22/11; Per NRC comments, modified stem to remove unnecessary information and reworded Choices "C" & "D" to improve discriminatory value of "D". - 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 will automatically be 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.

Question Misc. Info: LOIT, ISA-00-C, 2332B, I/A, S/A, MB-00607, NRC-2011

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

07/22/11; Per NRC comments, in Choice "B", changed ".. and automatically isolated .." to ".. and **will** automatically **be** isolated ..". - 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 sequential actions that the BOP must perform per EOP 2525, Standard Post Trip Actions?

☐ **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.

Question Misc. Info: MP2*LOIT, SW, 2326, RBCCW, 2330A, 2004; NRC-2011

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 Of 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.

09/02/11; per NRC comments, added "sequential" to the stem question statement. - rlc

NRC K/A System/E/A **System** 103 Containment System

Question #: 54	Question ID: 4000022	☑ RO	☐ SRO	☐ Student Handout?	☐ Lower Order?
	Rev. 1	☑ Selected for Exam		Origin: Bank	☐ Past NRC Exam?

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 #: 55	Question ID: 1100026	☑ RO	☐ SRO	☐ Student Handout?	☐ Lower Order?
	Rev. 1	☑ 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 1000 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** Auxiliary Recirculation Fans will have been manually started in slow speed.
All CAR Fans will have automatically started in slow speed.

☐ **B** Auxiliary Recirculation Fans will have been manually started in slow speed.
All CAR Fans will have been manually started in slow speed.

☐ **C** Auxiliary Recirculation Fans will not be running.
All CAR Fans will have been manually started in slow speed.

☑ **D** Auxiliary Recirculation Fans will not be running.
All CAR Fans will have automatically started in slow speed.

Question Misc. Info: MP2*LOIT* EOP 2532, LOCA, CTMT, CTMT Cooling, CAR, PIR, SIAS, NRC-2011

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. The CAR fans get a load shed from the sequencer on an LNP, and will be automatically started on sequence 1 (2 seconds after power is available).
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

07/22/11; per NRC comments, modified all four chioces to improve discriminatory value. - rlc

09/02/11; per NRC comments, modified choices 'B' and 'D' from "started **to** slow speed" to "started **in** slow speed". - rlc

09/16/11; per Exam Validation, corrected "Cut and Paste" error between choices "C" and "D" ("C" contains correct info, but "D" was originally designated as correct). - rlc

10/04/11; per NRC comments, changed PZR pressure given in stem from "**1410** psia" to "**1000** psia". - rlc

NRC K/A System/E/A System 103 Containment System

Question #: 55 | Question ID: 1100026 | ☑ RO ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

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

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 #: 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.

Question Misc. Info: MP2*LOIT*5658 [001 CED-01-C 2911], CEDS, CEAPDS, 2302, NRC-2011

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 <u>indicated</u> value of Tavg and what is the impact of this failure <u>without</u> 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.

Question Misc. Info: MP2*LORT*4397 [041 RRS-01-C 5130] (1/27/97) 2386, RRS, NRC-2011

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 <u>manually</u> deselected (procedurally directed action), the examinee may believe that the failed instrument is <u>automatically</u> 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 will 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.

Question Misc. Info: Incore Temperature Monitor System (ITM), NRC-2011

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.

07/22/11; Per NRC comments, in Choice "C", changed "**update**" to "**will update**", in Choice "D", changed "**not**" to "**NOT**" and removed all underline fonts. - 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 alarm on either of the particulate containment atmospheric monitors.

☐ **B** Containment atmosphere radiation triggering an alarm on the Kaman stack rad. monitor.

☐ **C** Enclosure Building differential pressure exceeds 0.5 inH2O as indicated on C-01.

☑ **D** Containment pressure on 2 or more wide range indications on C-01 exceeding 4.5 psig.

Question Misc. Info: MP2*LOIT, CPVIS, H2 Purge, 2314B, MB-02470, NRC-2008, NRC-2011

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 triggered by the CTMT High Range Radiation Monitors, 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 particulate atmospheric monitors would isolate the CTMT purge valves, especially when it does isolate the CTMT ventilation system 48" 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 trip setpoint for Fan-23, which could be exceeded if the EB were also being purged at the same time.

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.

07/22/11; Per NRC comments, reworded Choice "A" and clarified justification to better distinguish CTMT High Range Rad. Monitors from CTMT Atmosphere Particulate Rad. Monitors. Also, changed Choice "C" to a different parameter than that described in Choice "D" to make Choice "C" clearly wrong. - 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 #: 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 36 hours 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.

Question Misc. Info: MP2*LOIT, Loss of RBCCW to SFPC, NRC-2011

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.

Comments and Question Modification History

07/22/11; Per NRC comments, changed predicted restoration time in stem from "**4 to 6 days**" to "**36 hours**" to ensure RWST is capable of absorbing heat input of SFP Feed-and-Bleed cooling. - rlc

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, starting up following a refueling outage.
- All systems are in a 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 Emergency Generation Reduction, Turbine load was lowered more quickly then the Operator on the Steam Dumps could respond. While attempting to stabilize the plant,, the Operator on the Steam Dumps reported that S/G pressures were at 870 psia and rising. Steam and Feed flows were lowering.

Which of the following describes the impact on the stated parameter or calculated value, as compared to its value prior to the turbine load reduction?

☐ **A** Narrow Range Power will rise due to the lower density of the primary coolant.

☑ **B** Reactor Power will lower due to the rise in Reactor Coolant temperature.

☐ **C** Calorimetric power will rise due to the rise in Steam Generator Enthalpy.

☑ **D** Delta T Power will lower due to the rise in RCS Cold Leg temperature.

Accept B or D [initials] 11/4/11

Question Misc. Info: MP2*LOIT 2557, NRC-2011

Justification

B - CORRECT; Even though the core is at BOL conditions, at this power level MTC would still be negative. Therefore, rising S/G pressure and lowering Feed flow will result in rising RCS temperature, which will add negative reactivity causing power to lower.

D - Also CORRECT; Even though the reactor is at BOL conditions, at this power level, MTC would be negative. Therefore, as Tcold rises from the excessive "load reject", nuclear power will lower, resulting in Tcold rising faster than Thot (effectively - Delta-T lowers).
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.)

A - WRONG; A rise in RCS temperature will cause primary coolant density to lower; however, the negative MTC will overshadow the effects of changing density.
Plausible; The examinee may remember that rising water temperature causes density to lower. The lower density will result in increased leakage to the neutron detectors.

C - WRONG; If S/G temperature rises, then S/G enthalpy will also rise; however, feed flow is the biggest contributor to the calorimetric.
Plausible; If S/G temperature and pressure rise, then S/G enthalpy will also rise.

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

01/06/11; Reworded the question statement in the stem to clarify what was being asked, per comment from Sandy Doboe.
07/25/11; Per NRC comments: Changed question stem to "Turbine load" vs. "Generator output. Reworded stem and question such that it clearly indicates an operation resulting a rise in RCS temperature. Reworded all the Choices to ensure plausibility. - RJA
09/02/11; per NRC comments, modified stem question statement to clarify what the parameter values stated in the choices are being compared with. - rlc
09/19/11; per Exam Validation, to eliminate confusion as to whether choices refer to "actual" values or "indicated" values compared to actual values, each choice was modified to remove the word "indicate" and made grammatically correct based on removing it. - rlc

01/17/2011; During the exam review, it was noted that Choice "D" is also a correct answer. Credit was given for 2 correct answers.

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

Question Misc. Info: MP2*LOIT, Reactivity, Boron, Turbine, 2204, NRC-2011

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** Bypass the GS Regulator using 2-MS-182A, Bypass Feed MOV, and restore GS steam pressure to normal.
- ☑ **D** Open 2-EB-57, condenser air removal to Unit 2 stack isolation damper, and start one main exhaust fan.

Question Misc. Info: MP2*LOIT/LOUT, 2329, NRC-2011

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 CAR 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

07/22/11; Per NRC comments, reworded Choice "C" to improve symmetry and fixed typo on 2-MS-182**A**. Also deleted old "Change History" comments left over from previously copied question. - rlc

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. Take manual control of both Main Feed Pumps and maintain their speed constant.

☐ **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.

Question Misc. Info: LOIT, E25-00-C, 6.9Kv, Condensate, 2319A, 2525, MB-05431, NRC-2011

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
3. ARP-2590D-041, R0C1, Condensate Pump Trouble

Comments and Question Modification History

07/22/11; Per NRC comments, changed the second sentence in Choice "C" to match second sentence in Choice "A" to improve choice symmetry. Also added ARP-2590D-041, Condensate Pump Trouble, the references. - rlc

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 #: 65	Question ID: 1100031	☑ RO	☐ SRO	☐ Student Handout?	☐ Lower Order?
	Rev. 1	☑ 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 actions must be performed?

☑ **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** All Radioactive Liquid Waste Discharges must be secured during Mussel Cooking operations to ensure compliance with the station's NPDES permit.

Question Misc. Info: MP2*LOIT MB-00041 CWS-04-C, SWS, CWS, 2560, 2327, 2325D, NRC-2011

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 change 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

07/22/11; Per NRC comments, modified stem question statement to match syntax of the first three choices and modified the wording of Choice "D" to match the syntax of the other three choices. - 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

Question Misc. Info: MP2*LOIT, PPC, ESD, 2536, NRC-2011

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 small steam line break inside CTMT. (Other malfunctions are added to complicate the diagnosis.)

Comments and Question Modification History

12/17/10, Changed distractor B to an intersystem LOCA in the Letdown system. RJA

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 #: 67	Question ID: 53293	☑ RO ☐ SRO	☐ Student Handout?	☐ Lower Order?
	Rev. 4	☑ Selected for Exam	Origin: Bank	☐ Past NRC Exam?

A plant power ascension is in progress. The plant computer has calculated thermal power to be 1860 MWth and the operators are holding power steady at this point, temporarily, in order to perform SP-2601D, Power Range Safety Channel and Delta-T Power Channel Calibration.
The operator performing the surveillance notes that Nuclear Instrument System (NIS) power on each channel is as follows:

- Channel 'A' = 75%
- Channel 'B' = 73%
- Channel 'C' = 73%
- Channel 'D' = 74%

Based on the given conditions, which of the following actions is required per SP-2601D?

☐ **A** Perform the surveillance on Channels "A" and "D" before proceeding to Channels "B" and "C".

☑ **B** Notify Reactor Engineer of the power indications prior to performing any channel calibrations.

☐ **C** Request I&C verify the calorimetric accuracy prior to performing any channel calibrations.

☐ **D** Perform the surveillance on Channels "B" and "C" before proceeding to Channels "A" and "D".

Question Misc. Info: MP2*LORT*1936 [015 NIS-01-C 5111] (9/30/97) 2601D, 2380, 2203, RPS, NI, CALOR, NRC-2011

Justification

B - CORRECT; SP 2601D, R16C01, Step 4.1.6 states that if calorimetric power and NI power do not agree within 5%, notify RE prior to performing the surveillance. Thermal power is 1860/2700 x 100% = 68.89% and two of the indications are >/= 5% above this value.

A - WRONG; RE must be notified when NI power and calorimetric power disagree by >/= 5%.
Plausible; Channels "A" & "D" differ from the calorimetric by the greatest amount, therefore it would make sense to have I&C check the calorimetric calculation first.

C - WRONG; RE must be notified when NI power and calorimetric power disagree by >/= 5%.
Plausible; With the Nis differing from the calorimetric by >5%, it would make sense to have I&C verify their calibration.

D - WRONG; RE must be notified when NI power and calorimetric power disagree by >/= 5%.
Plausible; If it is believed that the highest, and therefore most conservative reading is the one to use, then the lowest channels should be calibrated first.

References

SP-2601D, Power Range Safety Channel and Delta-T Power Channel Calibration, R16C1, Step 4.1.6

Comments and Question Modification History

09/02/11; per NRC comments, reworded stem from "should" statement to "is required" statement. Also, modified choices to better fit "required actions" per procedure and reordered choices to make the correct answer "count" come out even. - rlc

09/28/11; per NRC comments, added "Power Range Safety Channel and Delta-T Power Channel Calibration" to the stem and the References for the question. - 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 2.1.23 **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 740 MWe and hydrogen pressure is 58 psig.

What is the maximum amount of <u>overexcitation</u> in MVARs that the generator can produce and stay within the limits of the curve?

☐ **A** 420 MVARs

☐ **B** 435 MVARs

☑ **C** 540 MVARs

☐ **D** 555 MVARs

Question Misc. Info: MP2 LOIT AOP 2580 Degraded Voltage, NRC-2011

Justification

C - CORRECT; Per AOP 2580, Attachment 1 (**required**), at 740 MWe and 58# hydrogen, the max MVAR loading is ~ 540 MVARs. Also, an "overexcited" generator would produce <u>lagging</u> MVARs.

A - WRONG; Unit 2 is required to have a "lagging" power factor (the generator is not allowed to operate "under excited").
Plausible; The "X" and "Y" axis equates to ~420 MVAR limit if an underexcited machine is considered.

B - WRONG; The generator can not operate under excited and the actual hydrogen pressure must be considered. Not what it normally is.
Plausible; A "leading" power factor equates to ~435 MVAR limit if the hydrogen pressure were 60 psig.

D - WRONG; The actual hydrogen pressure of 58 psig must be used on this curve, not the normal pressure of 60 psig.
Plausible; 60 psig and 740 MWe equates to a 555 MVAR limit lagging.

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.

07/25/11; Per NRC comments, modified stem from soliciting maximum "**lagging**" MVARs to soliciting maximum "**overexcitation**" in MVARS with a hydrogen perssure of 58 psig instead of the normal 60 psig and modified choices to match changes in the stem. - rlc

09/02/11; per NRC comments, changed choice "A" from 400 to 420 MVARs and choice "B" from 415 to 435 MVARs. Also added explanation of overexcitation of the generator to Justification. - 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 #: 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

Question Misc. Info: MP2*LOIT, SW, 2612A, MB-00112, NRC-2001, NRC-2002 [K/A 2.1.25], NRC-2011
Requires the use of SP 2612A,-003

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. 1 | ☑ 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. Next, a third operator will go out ALONE and verify the valve is properly locked in position.

☐ **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. Next, a third operator will go out ALONE and verify the valve is properly locked in position.

☐ **C** One operator will go out ALONE and fully close the valve, then reopen it one full turn and lock it in that position. Next, the second operator will go out ALONE and verify the valve is open and properly locked in position.

☐ **D** One operator will go out ALONE and fully close the valve, then reopen it one full turn. Next, the second operator will go out ALONE and verify the valve is properly positioned using system parameters, then lock it in that position.

Question Misc. Info: MP2*LORT*5613, PI-AA-500, ADMIN, NRC-2011

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

07/25/11; Per NRC comments, modified each choice to more clearly describe various ways of performing an **"Independent Verification" of a Locked Throttle Valve**. - 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 #: 71	Question ID: 1100061	☑ RO	☐ SRO	☐ Student Handout?	☐ Lower Order?
	Rev. 0	☑ Selected for Exam		Origin: New	☐ Past NRC Exam?

The plant is at 100% power, steady state, forcing Pressurizer Sprays for boron equalization.
Then, VR-21 is lost due to an internal bus fault and, after assessing the situation, the crew performed the required actions to stabilize the plant. It was noted that the 10 Minute Battery Backup for the Foxboro IA System immediately failed on loss of VR-21.
The following additional conditions now exist:

- Plant power = 100% and stable.
- Pressurizer pressure = 2217 psia and slowly rising.
- Pressurizer level = 60% and dropping very slowly.
- All electrical busses are energized with the exception of VR-21.

Which of the following LCOs must be entered due to these conditions?

☑ **A** 3.2.6 - DNB Margin

☐ **B** 3.4.4 - Pressurizer

☐ **C** 3.8.2.1 - Onsite Power

☐ **D** 3.5.2 - ECCS Subsystems

Question Misc. Info: MP2*LOIT, PLPCS, VR-21, NRC-2011

Justification

A - CORRECT; PZR pressure must be >2225 psia in this MODE to meet the DNB TS.

B - WRONG; The PZR no longer has a minimum level (only max. @ 70%) and only the Backup Heaters would be lost with a loss of VR-21.
Plausible; The BU heaters cannot be recovered without VR-21 and the level is below the normal setpoint by 5%, which is how much above the normal setpoint the TS limit is.

C - WRONG; VR-21 is not one of the TS control power supplies.
Plausible; VR-21 powers many of the control systems necessary for stable control of the plant during At Power and shutdown operation.

D - WRONG; The charging pumps do NOT need to automatically start to meet the requirements of this TS.
Plausible; Due to the conditions given, the charging pumps must be secured such that they will not start for any signal, emergency or otherwise.

References

TS 3.2.6, DNB Margin and TRM Appendix 8.1, COLR, section 2.7b, DNB Margin, Pressurizer Pressure.

Comments and Question Modification History

09/02/11; per NRC comments, replaced question with one based on original feedback from the Lead Examiner. - rlc

09/28/11; per NRC comments, added question references. - 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 #: 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 slow moving 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 high voltage in this region causes a current flow that exceeds the current generated by all ionizations.

☑ **D** Even with a constant voltage, the secondary ionizations in this region are unstable.

Question Misc. Info: MP2*LOIT*MB-03114, RM, 2383, Rad Monitor NRC-2011

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.; therefore, secondary ionizations are unstable and will NOT produce an output proportional to the radiation levels.

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

Comments and Question Modification History

07/26/11; Per NRC comments, reworded choices due to psychometric flaws.

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: 1100057 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant is operating in MODE 5. In accordance with OP 2207, Plant Cooldown, the crew has initiate maximum Additional Purification flow in preparation for an impending crud burst to be induced by plant chemistry.

Which of the following areas will see higher radiation levels during the clean up?

☐ **A** The running Charging Pump Rooms

☐ **B** The Volume Control Tank Room

☐ **C** The Clean Waste Tank Room

☑ **D** The "A" or "B" Safeguards Rooms

Question Misc. Info: MP2*LOIT, ALARA, CVCS, 2304, 2207, NRC-2011

Justification

D is correct; Additional Purification flow is from the discharge of the SDC Heat Exchangers through the Letdown Heat Exchanger, to the Letdown Ion Exchanger and back to the LPSI Pump suction.

A is wrong; Additional Purification does NOT go through the Charging Pumps.
Plausible: Excess Letdown does go through the Charging Pumps. The examinee may confuse Excess Letdown with Additional Purification.

B is wrong; The VCT Room will NOT necessarily see higher radiation levels because Additional Purification flow is isolated from the VCT. Flow is diverted back to the SDC System prior to entering the VCT.
Plausible: The examinee think that Additional Purification is through the VCT before returning to the LPSI Pump suctions.

C is wrong; The Clean Waste Tanks are NOT placed in service for the Additional Purification flow path.
Plausible: On additional Purification, there is NO provision for diverting flow to the Clean Waste System; however, flow may be diverted with the realignment of one valve. The examinee may feel that the more contaminated fluid from the RCS should be diverted to Rad Waste until radiation levels are lowered.

References

1. OP 2207, Plant Cooldown, Section 4.23.
2. Lesson Plan, CVC-00-C, Chemical and Volume Control System, Page 104.

Comments and Question Modification History

07/26/11; Per NRC comments: Unable to salvage question 1150023 without making the correct answer too obvious. Replaced question. - RJA

09/19/11; per Exam Validation, minor change to second sentence of stem to clarify that question is soliciting radiation level change due to the Excess Purification flow path and not the aspect of purification cleanup. - 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 2.3.14 **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 #: 74	Question ID: 153723	☑ RO	☐ SRO	☐ Student Handout?	☑ Lower Order?
	Rev. 0	☑ Selected for Exam		Origin: Bank	☐ Past NRC Exam?

A fire in Appendix "R" Fire Area R-1 has resulted in the evacuation of the Control Room.

The crew has just entered AOP 2579A, "Fire Procedure for Hot Standby Appendix R Fire Area R-1".

Which one of the following actions, per AOP 2579A, is required to be completed within the first 30 minutes of the Control Room evacuation?

☐ **A** Power is established to a vital 4160 Volt bus

☑ **B** Feed flow is established to a steam generator

☐ **C** RCS make up is established via a charging pump

☐ **D** "C" Battery Charger is aligned to Facility 2

Question Misc. Info: MP2*LOUT, Fire, 2579, NRC-2002, NRC-2011

Justification

B is correct. The caution prior to step 1 of AOP 2579A states, "Failure to initiate Auxiliary Feedwater flow to any SG within 30 minutes of a loss of normal feedwater may result in that SG boiling dry.

A is incorrect. Power must be restored within 4 hours of the reactor shutdown.
Plausible; Power is required to utilize the electric AFW pumps, which are normally used. However, lack of power should not delay feeding the S/Gs because the Turbine Driven AFW pump is assumed available.

C is incorrect. Charging flow is required to be restored within 4 hours of the reactor trip.
Plausible; Charging pump restoration is a limiting requirement on a loss of all AC power.
D is incorrect. "C" battery Charger is required to be aligned to Facility 2 prior to depletion of the "B" Battery. This is assumed to take longer than 30 minutes.
Plausible; This is a requirement that must be done expeditiously, but not in under 30 minutes.

References

AOP 2579A, Fire Procedure for Hot Standby Fire Area R-1

Comments and Question Modification History

07/26/11; Per NRC comments, question replaced to improve K/A match.

09/02/11; per NRC comment, removed the word "Auxiliary" from the correct answer. - rlc

09/28/11; per NRC comments, added reference. - 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 2.4.25 **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 failed to transfer to the RSST and 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.

Question Misc. Info: MP2*LOIT, VR-21, 2504B, NRC-2011

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. On a Loss Of Offsite Power (failure of 24C to transfer to the RSST) with a concurrent SIAS (caused by the ESD in CTMT), the operators are not allowed to re-start the vital IAC and are required to cross-tie air with Unit 3.

A - WRONG; Although the #2 AFRV will fail open on loss of DC, the #1 AFRV can still be closed to prevent feeding the break.
Plausible; Loss of 201B de-energizes half of the vital DC busses and if the "B" steam header ruptured the pump would have to be tripped

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

07/26/11; Added reason for why 24C was being powered by the EDG (faillure to transfer to RSST). Also, added explanation to justification as to why SIAS was actuated. - rlc

09/02/11; per NRC comment, reworded Justification for choice "A" to match changes made due to previous feedback. - 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. 1	☑ 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.
- Six CEAs are stuck out.
- Bus 24C is faulted.
- Facility 2 SIAS, CIAS, EBFAS, and MSI verified fully actuated with all components functioning as designed.
- All other electrical buses are energized.
- "B" & "D" RCPs are operating.
- Pressurizer level is 10% and NOT restoring.
- Reactor Vessel Head level is 100% and stable.
- All available charging pumps are operating, but charging flow is 'zero'.
- Pressurizer pressure is 1550 psia, and slowly lowering.
- SG levels are 45% and stable.
- SG pressures are 860 psia and stable.
- CETs are 532° F and stable.
- Containment pressure is 4 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.
- Radiation monitors inside Containment are rising slowly.

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 after reevaluating plant conditions?

☐ **A** Immediately transition to EOP 2530, Station Blackout.

☐ **B** Immediately transition to EOP 2540A, Functional Recovery of Reactivity Control.

☐ **C** Immediately transition to EOP 2532, Loss of Coolant Accident.

☑ **D** Immediately transition to EOP 2540, Functional Recovery.

Question Misc. Info: MP2*LOIT, EOP, 2525, 2532, LBLOCA, NRC-2011, 55.43(b)(5)

Justification

D is correct. Reactivity Control is not being met due to the stuck CEAs and lack of any boron injection. When using the Diagnostic Flow Chart, if the Reactivity 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 the Diagnostic Flow Chart says to consider EOP 2530 under the existing conditions, the US must recognize that all conditions for a Station Blackout do not exist.
Plausible: The examinee may believe that the loss of Vital DC would result in a loss of the only available Vital AC buss.

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 six stuck CEAs. Because Reactivity Control is the highest safety function, the examinee may believe that it should be addressed immediately.

C is incorrect. Reactivity Control is affected, requiring the crew to immediately address this Safety Function through the Functional Recovery procedure.
Plausible: Inventory Control is in jeopardy due to the Small Break LOCA. The examinee may recognize that EOP 2532 mitigating strategy will direct the crew to cool down and depressurize the RCS, which would then allow SI flow to occur and meet the Reactivity Control Safety Function. However, procedure usage requires the higher Safety Function be addressed immediately per its applicable procedure.

References

OP 2260, EOP Users Guide

Comments and Question Modification History

06/27/11; Per NRC, modified Stem from **"Two CEAs are stuck out"** to **"Six CEAs are stuck out."** - rlc
06/27/11; Per NRC, modified all choices to **remove all text before the word "immediately"**. - rlc

Question #: 76 **Question ID:** 1100034 ☐ RO ☑ SRO ☐ Student Handout? ☐ Lower Order?

Rev. 1 ☑ Selected for Exam **Origin:** New ☐ Past NRC Exam?

08/03/11; Per Operation Management comments, modified given plant conditions to ensure only one correct answer. - rlc
09/05/11; Modified "PZR pressure" rate of change in stem from **"continuing to lower"** to **"slowly lowering"** to improve clarity. - 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 2.4.6 **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. 2	☑ Selected for Exam		Origin: New	☐ Past NRC Exam?

A plant heatup has just been started per OP 2201 and the following conditions presently exist:

- RCS Temperature is at 210°F and slowly rising.
- RCS pressure is stable at the minimum allowed for "A" and "B" RCP operation.
- "A" and "B" RCPs have just been started.
- Shutdown Cooling (SDC) has just been secured.
- "C" and "D" RCP breakers have just been racked up.

Then, "A" RBCCW Header flow is lost when the "A" RBCCW Heat Exchanger outlet valve fails closed. As flow is restored to the "A" RBCCW Header, the following indications are seen for the "A" RCP:

- Annunciator C-02/3, AB-17, "RCP A STR TEMP HI" alarms.
- Motor Stator Temperature is noted as 270°F and slowly rising.

Which of the following choices contains a correct sequence of actions to be directed by the US, per the applicable procedures?

☐ **A**
1. Per OP 2201, Attachment 6, Contingency Actions, raise RCS pressure as required.
2. Per OP 2301C, Reactor Coolant Pump Operation, start the "C" and "D" RCPs.
3. Per OP 2201, Attachment 6, Contingency Actions, secure the "A" and "B" RCPs.

☑ **B**
1. Per ARP 2590B-066, AB-17 "RCP A STR TEMP HI",secure "A" RCP.
2. Per OP 2201, Attachment 6, Contingency Actions, secure the "B" RCP and raise RCS pressure as required.
3. Per OP 2301C, Reactor Coolant Pump Operation, start "C" and "D" RCPs.

☐ **C**
1. Per ARP 2590B-066, AB-17 "RCP A STR TEMP HI",secure "A" RCP.
2. Per OP 2310, Shutdown Cooling Operation, place SDC in Intermittent Operation.
3. Per OP 2201, Attachment 6, Contingency Actions, secure the "B" RCP and lower RCS pressure as required.

☐ **D**
1. Per OP 2301C, Reactor Coolant Pump Operation, start "C" and "D" RCPs.
2. Per AOP 2564, Loss of RBCCW, secure the "A" RCP.
3. Per OP 2201, Attachment 6, Contingency Actions, secure the "B" RCP.

Question Misc. Info: MP2*LOIT RCP, OP 2301C, NRC-2011, 55.43(b)(5)

Justification

B - 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. Exceeding the stator temperature limit of 260°F requires the "A" RCP be immediately secured, even if it involves a plant trip from 100% power. In addition, the minimum NPSH requirements for "A" & "B" RCP operation is based on both pumps running. Therefore, "B" RCP is not allowed to operate alone and must be immediately secured when "A" RCP is secured. Although "C" & "D" RCPs are available to start, the minimum NPSH for "C" & "D" RCPs is higher than that for "A" & "B" RCPs. Therefore, pressure must first be raised before they can be started. The SRO is expected to know that even though there will be NO Tech. Spec. required RCS flow for a short period of time, this is the procedural required course of action for the given plant conditions.

A - WRONG: Even though tripping the RCPs will cause a loss of Tech. Spec. required RCS flow with unstable temperatures, ARP 2590B-066 requires the RCP be immediately secured. There is no allowance to wait for pressure to be raised and other RCPs to be started before securing the over heating RCP.
Plausible; The examinee may believe that running any RCP is better than no RCS flow given these plant conditions.

C - WRONG: Tripping both RCPs cannot be delayed until SDC can be restored as there is no guidance for single RCP operation.
Plausible; This would be an acceptable action if MP2 were allowed to operate a single RCP at any time other than starting the first one. The examinee may believe that single pump operation may be allowed in MODE 4 as the RCPs are being used for RCS heatup.

D - WRONG: The minimum NPSH requirements for the "A" and "B" RCPs is lower than that required for the "C" & "D" RCPs. Therefore, pressure must be raised before these two pumps can be started.
Plausible; The examinee may believe that starting "C" & "D" RCPs would be acceptable if "A" & "B" were allowed to run.

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.
3. OP 2201, Plant Heatup, Att. 6, Step 3, Contingency actions for loss of 1 RCP when 2 were running and SDC is not in service.

Question #: 77	Question ID: 1100035	☐ RO	☑ SRO	☐ Student Handout?	☐ Lower Order?
	Rev. 2	☑ Selected for Exam	Origin: New		☐ Past NRC Exam?

Comments and Question Modification History

06/27/11; Per NRC, modified stem question to "**must be given by the US, per the applicable procedures,** " - rlc

09/02/11; per NRC comments, removed the word "Immediately" from the beginning of each choice.
Did NOT reword stem question statement to focus on specific AOP because the Loss of RBCCW procedure, AOP 2564, covers only the need to secure the "A" RCP and not the actions necessary to deal with required RCP combinations at the specified plant conditions. OP 2201, Plant Heatup, Attachment 6, Contingency Actions, Step 3, Loss of an RCP with SDC out of service covers the remaining actions that must be taken. - rlc

10/04/11; per NRC comments, reworded each choice to meet the 55.43(b)(5) guidelines. - rlc

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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. 2	☑ 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.
- The crew has transitioned to the Event Specific EOP.

Which of the following actions must the US direct during the performance of the applicable EOP, and what is the reason for this action?

☑ **A** Per EOP 2534, Steam Generator Tube Rupture, have a PEO manually close the Turbine Driven Aux Feedwater Pump steam supply valve, MS-202, to prevent the loss of a barrier and escalation of the event classification.

☐ **B** Per OP 2325A, Circulating Water System, have the BOP cross-tie condenser water boxes, to ensure condenser steam dump valve availability and minimize the radiation release to the environment while cooling down and isolating the affected S/G.

☐ **C** Per EOP 2534, Steam Generator Tube Rupture, have a PEO isolate Hotwell Reject to stop the potential overflow of the Condensate Storage Tank, and prevent the loss of a barrier and escalation of the event classification.

☐ **D** Per EOP 2541, Appendix 23. Restoring Electrical Power, have the BOP cross-tie 480 VAC busses, to maintain condenser steam dump valve control power and minimize the radiation release to the environment while cooling down and isolating the affected S/G.

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Question Misc. Info: MP2*LOUT, AFW, 2534, 2322, SGTR, MB-04750, NRC-2011, 55.43(b)(5)

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; Condenser water boxes are not required to be cross-tied to maintain a vacuum when 2 circ. pumps are lost.
Plausible: Condenser water boxes are cross-tied by procedure (OP 2325A) during a plant shutdown to ensure even loading of the condenser and cooling of the main turbine rotor, but there is no procedure requirement to do this during EOP use.

C - WRONG; The loss of bus 24D does not result in an overflow of the Condensate Storage Tank.
Plausible: If the examinee believes that the loss of Bus 24B (due to loss of 24D) will cause the hotwell to continue to fill and reject to the Condensate Storage Tank, which it cannot do.

D - WRONG; Loss of Facility 2 power will put VR-21 on its UPS, but its battery is designed to last long enough (one hour) to allow for cooling down and isolating a ruptured S/G, which is required to be accomplished in one hour.
Plausible: The examinee may remember the old VR-21 control power battery backup of only 10 minutes. The VR-21 UPS modification was completed during the last refueling outage (2R20).

References

TG EOP 2534, Step 14 (#2 SG), AOP 2503F, Load List.
EAL Basis Document.

Comments and Question Modification History

06/28/11; Per NRC, **reworded choices 'B' & 'D'** with minor word change to question sentence in stem. - rlc

09/02/11; per NRC comments, reworded valve MS-202 name to match EOP wording (not OP-2322 as suggested) and corrected minor typos in stem and choice "A". - rlc

09/16/11; per Exam Validation, modified choice "C" to be incorrect. Under the given conditions, hotwell could possibly reject to "Surge" tank, but not the "Storage" tank. - rlc

09/28/11; per NRC comments, reordered wording of choice "A" to more closely fit actual name of MS-202. - rlc

10/04/11; per NRC comments, reworded each choice to meet the 55.43(b)(5) guidelines. - rlc

Question #: 78 | Question ID: 1190004 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

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

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 #: 79	Question ID: 1100037	☐ RO	☑ SRO	☐ Student Handout?	☐ Lower Order?
	Rev. 1	☑ 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, multiple alarms are received. After a brief scan, the board operators report the following:

- #2 FRV Bypass Valve is closed.
- #1 and #2 SG Narrow Range Level indication (LI-1113B and LI-1123B, respectively) are deenergized.
- Pressure in SITs 1 - 4 indicate: 210 psig, 0 psig, 215 psig, 220 psig, respectively.
- RCS temperature is 531°F and slowly rising.
- RCS pressure is 2235 psia and lowering.
- Annunciator on C01, B-38, "ACTUATION CAB 6 POWER SUPPLY TROUBLE" is in alarm.
- Annunciator on C01, B-27, "STM. GEN. PRES. LO LO B" is in alarm.

The US directs the RO and BOP to stabilize the plant per the appropriate AOP.

Which of the following states the administrative implications and applicable requirements under these conditions?

☐ **A** Required facilities of ESAS are inoperable.
Restore the inoperable facility of ESAS to OPERABLE status within 7 hours or be in HOT SHUTDOWN within the next 6 hours.

☐ **B** Required facilities of MSI are inoperable.
Restore the inoperable facility of MSI to OPERABLE status within 24 hours or be in COLD SHUTDOWN within the next 36 hours.

☑ **C** Required facilities of vital power are inoperable.
Restore the inoperable facility of vital power to OPERABLE status within 8 hours or be in COLD SHUTDOWN within the next 36 hours.

☐ **D** Required facilities of SIT indication are inoperable.
Restore the inoperable facility of SIT indication to OPERABLE status within 24 hours or be in HOT SHUTDOWN within the next 6 hours.

Question Misc. Info: MP2*LOIT/LOUT, SRO, VIAC, AOP 2504D, VA-20, TSAS, Tech Spec, MB-05743, NRC-2011, 55.43(b)(2)

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. is based on the AC Electrical Power Distribution.

A is incorrect. While this would cause a loss of Facility 2 ESAS, it is NOT necessary to log into TSAS 3.0.3 because TSAS 3.3.2, Action 5 covers this condition.
Plausible: The examinee may believe that the loss of Facility 2 ESAS would require entering and following TSAS 3.0.3 because there is no Tech Spec Action for one whole facility of ESAS to be inoperable.

B is incorrect. MSI is inoperable, however the time specified is NOT the required time for this condition..
Plausible: Loss of VA-20 does prevent automatic isolation of Main Feed Water on an MSI actuation (either facility). The Tech Spec Action is plausible in that it is correct for a loss of the SIT indication, which VA-20 powers.

D is incorrect. Although #2 SIT indications would be lost on a loss of VA-20, this is the wrong TSAS for this condition.
Plausible: The examinee may NOT remember which TSAS for the SITs is applicable for loss of indications. The Tech Spec Action is correct for a loss of the SIT tank for other than loss of boron or indication.

References

AOP 2504D
TS 3.8.2.1

Comments and Question Modification History

07/01/11; Per NRC, modified the stem to solicit dominant administrative implication of the lost power supply. Also, modified choices to only solicit administrative effect of the lost VIAC power supply, to improve plausibility of distracters and SRO level. - rlc

09/02/11; per NRC comments, in each choice, changed the word **"bus"** to **"facility of $$$"**. Where **"$$$"** is the item of focus in the first sentence of each distracter. Also modified stem to remove direct information on specific safety channel lost and gave indications suggested by NRC examiner. - rlc

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

Question #: 79 | Question ID: 1100037 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

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

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

The plant is operating normally in MODE 1 at 100% power with the following conditions:
- "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 and the crew has successfully completed the actions of AOP 2565, Loss of Service Water.

Which of the following describes the status of the Service Water System based on the actions taken in AOP 2565?

☐ **A** Tech. Spec. 3.7.4.1, Service Water System, is met. Tech. Spec 3.0.5 must be entered.

☑ **B** Tech. Spec. 3.7.4.1, Service Water System, is not met. Entry into Tech. Spec 3.0.5 is not required.

☐ **C** Tech. Spec. 3.7.4.1, Service Water System, is met. Entry into Tech. Spec 3.0.5 is not required.

☐ **D** Tech. Spec. 3.7.4.1, Service Water System, is not met. Tech. Spec 3.0.5 must be entered.

Question Misc. Info: MP2*LOIT, 2565, SW, Service Water, AOP, NRC-2011, 55.43(b)(2)

Justification

B is correct. When the "A" Service Water Pumps trips, the "B" Service Water is started to restore flow to the Facility 1 SW header, per AOP 2565 guidance. The AOP goes on to direct the US to evaluate applicability of Service Water System Tech Spec 3.7.4.1. The AOP actions required to start the "B" Service Water Pump would prevent it from starting on its emergency power supply. Therefore, it no longer meets the TS requirements and the Facility 1 SW header is inoperable. However, actions required by TS 3.0.5 do not apply because there is no Facility 2 Tech. Spec. equipment inoperable.

A is incorrect. Tech Spec 3.7.4.1 is NOT met with the "B" Service Water Pump running while powered from the opposite facility.
Plausible: Because TS 3.0.5 could be more restrictive that TS 3.7.4.1, the examinee may believe the loss of the emergency power supply to the SW pump is the overriding consideration.

C is incorrect. Although the "B" SW pump has restored flow to the Facility 1 SW header, the header is inoperable because the pump has no emergency power supply.
Plausible: If the examinee does not integrate the specific actions of the AOP (start the pump, do NOT swap power supplies) with their impact on given facility alignment, this action would seem acceptable.

D is incorrect. Actions required by TS 3.0.5 do not apply because there is no Facility 2 Tech. Spec. equipment inoperable.
Plausible: The examinee could justify this action if the potential unavailability of the "A" EDG, due to the inoperability of the "A" SW header, were used to justify taking the actions required by TS 3.0.5.

References

AOP 2565, Loss of Service Water, St. 4.4, TS 3.7.4.1 and TS 3.0.5

Comments and Question Modification History

09/02/11; per NRC comments, modified each choice to better align with required wording of TS. - 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: 9000012	☐ RO	☑ SRO	☐ Student Handout?	☑ Lower Order?
	Rev. 1	☑ Selected for Exam		Origin: Bank	☑ Past NRC Exam?

The plant is in MODE 6 with the following conditions:

- Fuel movement is in progress.
- The Personnel Airlock Doors are open
- The Equipment Hatch is open.
- Containment Purge is in operation.
- Containment Atmosphere Radiation Monitor, RM-8123, is out of service for repairs.

The Auxiliary Building PEO has just reported that the blower for Containment Atmosphere Radiation Monitor, RM-8262, has tripped and cannot be restarted.

Which of the following actions must be taken and why?

☐ **A** Immediately suspend CORE ALTERATIONS and establish Containment Closure prior to resuming fuel movement, to ensure a potential fuel handling accident in Containment is NOT released to the environment.

☐ **B** Immediately suspend CORE ALTERATIONS and restore the Radiation Monitor blower prior to resuming fuel movement, to ensure a potential fuel handling accident in Containment is NOT released to the environment.

☑ **C** Ensure a control room operator is specifically assigned to close the Containment Purge Valves within 30 minutes of an event, to ensure Containment Closure is reestablished in case of a fuel handling accident in Containment.

☐ **D** Restore the Containment Purge Valves to OPERABLE status within the next 30 minutes or immediately close the Purge Valves, to ensure Containment Closure is reestablished in case of a fuel handling accident in Containment.

Question Misc. Info: MP2*LOUT, Purge, 2314B, TS, MB-06206, NRC-2009, NRC-2011, 55.43(b)(2)

Justification

C IS CORRECT; TS 3.9.4 requires that Containment Purge Valves either be closed by an automatic isolation or be capable of being closed under administrative control. A specific individual must be designated as available to close the Purge Valves within 30 minutes of a fuel handling accident in Containment.

A is incorrect; CORE ALTERATIONS do NOT need to be suspended and Containment Closure is still available.
Plausible if the examinee believes that the Purge Valves need to be closed by an automatic isolation signal. (Only one Containment Radiation Monitor needs to be OPERABLE to initiate and automatic closure of all 4 Purge valves.) The examinee may also believe that the loss of the only remaining Radiation Monitor (and automatic isolation of the Purge Valves) results in a loss of Containment Closure. (Containment Closure must be set or available during CORE ALTERATIONS.)

B is incorrect; CORE ALTERATIONS do NOT need to be suspended; however, it would be appropriate to have the Radiation Monitor blower repaired.
Plausible if the examinee believes that the Purge Valves need to be closed by an automatic isolation signal.

D is incorrect. In MODE 6, the Purge Valves are still considered OPERABLE even if they are NOT able to be closed by an automatic isolation signal.
Plausible because Tech Spec 3.6.3.1 requires each Containment Isolation Valve to be OPERABLE (in MODES 1, 2, 3, and 4). These valves are demonstrated OPERABLE by verifying the automatic signal functions or the valves are closed and secured. This Spec does NOT apply to the Containment Purge Valves in MODE 6.

References

Tech. Spec. 3.9.4 LCO; Containment Penetrations

Comments and Question Modification History

09/28/11; per NRC comments, question replaces Q#1100062 due to excessive overlap of original question with Q#90. Minor wording change to the stem question statement to improve sentence structure and include applicable procedure by name. Also reordered the choices to make "A" correct and even the count of correct answers. - rlc
Note: Original question was linked to K/A 065/AA2.06 on SRO-U exam NRC-2009.
09/29/11; per NRC comments, modified choices "B", "C" & "D" to eliminate overlap with Q#53. Also corrected Justifications for these choices. - rlc

10/05/11; Per NRC comments, selected new K/A due to original K/A not lending to an SRO level question. Selected question per new K/A that meet SRO discriminatory requirements. - rlc

Question #: 81	Question ID: 9000012	☐ RO	☑ SRO	☐ Student Handout?	☑ Lower Order?
	Rev. 1	☑ Selected for Exam		Origin: Bank	☑ Past NRC Exam?

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 #: **82** **Question ID:** 1100039 ☐ RO ☑ SRO ☐ Student Handout? ☑ Lower Order?

Rev. 2 ☑ 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.

Complete the following statement to describe the basis for the interlock that the RO must bypass to recover CEA #41.

The triggered interlock stops CEA movement, because CEA movement could ______________________.

☐ **A** further distort core flux tilt beyond that assumed for the LSSS setpoint determination.

☐ **B** degrade Shutdown Margin below that assumed in the safety analysis.

☑ **C** amplify localized core power distortions beyond that assumed in the safety analysis.

☐ **D** result in uneven fuel burnup beyond that assumed for the LSSS setpoint determination.

Question Misc. Info: MP2*LORT*5632 2556, TS, NRC-2011, 55.43(b)(2)

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 basis described is for the CEA Withdrawal Prohibit (CWP) interlock, 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; The basis described is for 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; The basis described is for the Long Term Steady State Insertion Limit, which has no interlock function when violated. Plausible; If the CEA were below the LTSSIL, continued operation at this level would result in unanalyzed fuel burnup.

References

Tech. Spec. Bases for 3.1.3, Moveable Control Assemblies.

Comments and Question Modification History

07/27/11; Per NRC comment, DISAGREE. The interlock triggered by the abnormal CEA alignment is a CEA Motion Inhibit (CMI), which is designed to stop further insertion and aggravate the already abnormal flux distribution. The CMI interlock stops all CEA motion, both insertion and withdrawal (which would also have the potential to aggravate an abnormal flux pattern). In addition, the question is soliciting knowledge of the Tech. Spec. Basis for the limit on CEA insertion (3.1.3.1), which is an SRO concept. While ROs may have system knowledge of the CEA Interlocks, it is not required that ROs have Tech. Spec. Basis knowledge. (Note: question reworded slightly to improve clarity.) - rlc

08/01/11; Per NRC comments, changed all choices to state that the triggered interlock stops "**all CEA movement**". - rlc

09/20/11; per NRC comments, revised stem question statement and all four choices to the wording suggested by Lead Examiner. Also made slight modification to the Justification of each choice to grammatically align with the applicable choice wording. - rlc

09/28/11; per NRC comments, added "the" to choice "D". - rlc

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. 2 | ☑ 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 one (1) of the six (6) fire detection instruments in the room appears to have failed, but the fire suppression system otherwise appears operational.

Which of the following actions, if any, must the US now take, based on TRM 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 hourly Fire Watch in the room within the next 2 hours.

☐ **C** Direct the on-watch PEO to check the room twice as often as dictated by the normal rounds.

☐ **D** No action is required with only one of the six fire detection instruments in the room failed.

Question Misc. Info: MP2*LOIT TRM, Fire Systems, Fire Watch, NRC-2011, 55.43(b)(1) & (2)

Justification

A - CORRECT; While ROs are responsible for 1 hour (or less) actions, the SRO must determine how many detectors are required to be OPERABLE. If the number of detectors is less than the minimum, then, 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 (not 3 hours after discovery), 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; The basis for TRM 3.3.3.7 states, "If a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the area is required to provide detection capability until the instrumentation is returned to OPERABILITY."
Plausible; The PEOs are all Fire Watch and Fire Brigade qualified. Therefore, the examinee may believe that doubling the PEO rounds frequency meets the requirement of frequent fire patrols.

D - WRONG; The TRM requires action to be taken with only one instrument failed.
Plausible; Implies required actions are similar to that of other instruments covered by the TS or TRM that do not directly affect core safety.

References

TRM 3.3.3.7, Action a.

Comments and Question Modification History

09/28/11; per NRC comments, added "if any" to the stem and changed choice "D" to "No action is required . ..". - 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. 2	☑ Selected for Exam		Origin: New	☐ Past NRC Exam?

The plant has tripped from 100% on a trip of the Main Turbine and the following conditions exist:

- #1 PORV, RC-402, is stuck partially open (dual indication).
- Bus 24C de-energized due to a bus fault.
- 24E is aligned to 24C.
- Pressurizer pressure = 1300 psia and stable.
- Pressurizer level = 75% and rising.
- Reactor Vessel Level on both channels (RVLMS) = 12%.
- Average CET temperature = 577°F.
- The crew has completed EOP 2525, Standard Post Trip Actions.

Which of the following actions must the US direct, per EOP 2532, Loss Of Coolant Accident, to mitigate the existing conditions?

☐ **A** Per Step 5, Optimize Safety Injection, reduce RCS pressure using main or auxiliary spray to raise safety injection flow.

☑ **B** Per Step 17, Perform Controlled Cooldown, initiate a controlled cooldown using the main steam dump valves to the condenser.

☐ **C** Per Appendix 23, Restoration of Electrical Power, isolate bus 24E from bus 24C and energize it from Unit 3, then start the "B" HPSI pump.

☐ **D** Per Appendix 24, Void Elimination, start all available CEDM cooling fans with RBCCW flow, to eliminate the head void.

Question Misc. Info: MP2*LOIT, ICCS, CET, SCM, 2387, MB-05109, NRC-2011, 55.43(b)(5)
Requires use of Steam Tables

Justification

B - CORRECT; The given conditions indicate inadequate heat removal due to a saturated RCS with vessel level below 43%. EOP 2532 gives guidance to commence an RCS cooldown (reflux cooling at this vessel level), which would lower pressure and raise SI flow.

A - WRONG; With a partially open PORV (and no other indicated break) lowering PZR pressure by spray flow would move more of the RCS inventory into the PZR and out the open PORV. With a SB-LOCA, RCS pressure, when at saturation, would be a function of the hottest source. At this time, that would be the core. Lowering pressure in the PZR would simply cause more steam generation in the vessel.
Plausible; Lowering RCS pressure by spray flow is a directed action in EOP 2532 to increase SI flow and regain control of RCS inventory and heat removal.

C - WRONG; Additional HPSI pumps would not help because the HPSI pumps are in parallel. Therefore, starting an additional pump would not raise HPSI discharge pressure above the existing RCS pressure, which at this time is above HPSI shutoff head.
Plausible; Restoring power to a dead vital bus and recovering SI pumps is directed by EOP 2532 to help regain control of RCS inventory.

D - WRONG; To balance RBCCW header loads, all CEDM coolers have been aligned to only the Fac. 1 RBCCW header. Therefore, none of the CEDM coolers can have RBCCW flow due to the loss of Fac. 1 vital power.
Plausible; EOP 2532 does not direct starting the CEDM coolers for this reason (to help eliminate a head void).

References

EOP 2541, Appendix 2, R2; RCS P/T Requirements.
Mitigating Core Damage LP (MCD-00-C) section on "Void Formation".

Comments and Question Modification History

07/12/11; Per NRC comments, rewrote question to make it SRO level by soliciting the required mitigating actions. - rlc

09/05/11; per NRC comments, modified choice "B" from ADVs to CDVs. Added information to the Justification for choices "A", "B" and "C". - rlc

10/04/11; per NRC comments, reworded each choice to meet the 55.43(b)(5) guidelines. - rlc

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. 3 | ☑ 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 EOP 2525, Standard Post Trip Actions, the crew noted that MFW, Condensate, and AFW are UNAVAILABLE, and have transitioned to EOP 2540, Functional Recovery procedures.

The following conditions now exist:

- RCS pressure is 1550 psia and slowly dropping.
- #1 SG level = 35% and stable.
- #2 SG level = 90" and dropping.
- #1 ADV open 50%.
- #2 ADV open 50%.

Which of the following actions are required to successfully mitigate these conditions, and why?

☐ **A** Per EOP 2540C1, Recovery of RCS Inventory, IC-2, Safety Injection, open BOTH PORVs, verify adequate Safety Injection flow, and open ONLY #2 ADV to 100%, to maintain CTMT Integrity and minimize the release to the environment.

☐ **B** Per EOP 2540D, Recovery of Heat Removal, HR-1, Appendix 12, SGTR Response, initiate an RCS cooldown NOT to exceed 80 °F/hr. and open BOTH PORVs, to ensure the Reactor Vessel belt line does NOT exceed design parameters.

☐ **C** Per EOP 2540C2, Recovery of RCS Pressure Control, PC-3, PORVs, open BOTH ADVs to cooldown the RCS at the maximum rate, then open BOTH PORVs at the 200°F subcool line, to prevent PTS of the RCS and Reactor Vessel.

☑ **D** Per EOP 2540D, Recovery of Heat Removal, HR-3, Once-Through-Cooling, open BOTH ADVs 100%, verify adequate Safety Injection flow, and open BOTH PORVs, to ensure core damage does NOT occur due to inadequate Heat Removal.

Question Misc. Info: MP2, TG2540D, EOP 2540D, NRC-2005, NRC-2011, 55.43(b)(5)

Justification

D - CORRECT; The SRO must recognize that even though the SGTR not being isolated is causing the loss of a Safety Function (CTMT Integrity) a higher level Safety Function (RCS Heat Removal) is also not being met. Therefore, both ADVs, as well as both PORVs, must be opened fully to initiate once-through cooling, or the limited PORV flow capacity will result in eventual core uncovery and fuel damage.

A - WRONG: Both ADVs must be utilized, even with a SGTR, based on the analysis for the OTC success path.
VALID DISTRACTOR: Applicant may assume #1 SG must remain isolated to minimize the radiation release, as required by other Safety Functions (RCS Inventory or Containment Integrity) of the Functional Recovery Procedures, especially with the possibility of AFW being restored soon.

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

Comments and Question Modification History

07/12/11; Per NRC comments, modified stem, choices and justification to improve plausibility and clarify SRO level. - rlc

08/01/11; Per NRC comment, expanded justification to improve understanding of SRO required knowledge. - rlc

09/05/11; per NRC comments, modified stem question statement, 2nd part, to simply state "and why", to better align with the choices given. - rlc

10/04/11; per NRC comments, reworded each choice to meet the 55.43(b)(5) guidelines. - rlc

NRC K/A System/E/A **System** E09 Functional Recovery

Generic K/A Selected

Question #: 85	Question ID: 1150008	☐ RO	☑ SRO	☐ Student Handout?		☐ Lower Order?
	Rev. 3	☑ Selected for Exam		Origin:	Mod	☐ 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 #: 86	Question ID: 1100043	☐ RO	☑ SRO	☐ Student Handout?	☑ Lower Order?
	Rev. 2	☑ Selected for Exam		Origin: New	☐ Past NRC Exam?

What is the basis for the MODE 1 through 4 RWST Boron Concentration Tech. Spec. LCO and what action is the US required to take if the concentration is found to be below the required limit?

☑ **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 required Shutdown Margin will be maintained following any transient causing an RCS cooldown, using the RWST as the sole source of borated water and without crediting for Xenon.
Prior to performing an RCS cooldown, verify at least one BASTs is operable, using SP 2601A, Borated Water Sources Verification, or cool down to MODE 5 within the next 36 hrs.

☐ **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 required Shutdown Margin will be achieved 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.

Question Misc. Info: MP2*LOIT*RWST volume and boron bases, NRC-2011, 55.43(b)(2)

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; This is a mix of the higher and lower mode bases, but is not totally inclusive of either.
Plausible; The examinee may partially remember the basis for the boron concentration requirements for the various borated sources.

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 from 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

Comments and Question Modification History

09/16/11; per NRC comments, modified choice "D" to improve plausibility. - rlc

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. 3	☑ Selected for Exam		Origin: Bank	☐ Past NRC Exam?

The unit is at 100% power.
The "A" EDG was taken out of service yesterday and is scheduled to remain out of service for the next 8 days to complete on-line maintenance in accordance with TSAS 3.8.1.1.b.4. All the requirements of this ACTION have been met.

Then, the Turbine Building PEO finds the Turbine Driven Auxiliary Feed Pump Steam Supply valve, MS-464 (SV-4188), in the "tripped" position and cannot relatch it. Maintenance investigates and reports the steam inlet valve latch is broken and it must be repaired before the pump can be operated.

Which one of the following describes the action required based on the applicable Technical Specifications?

☐ **A** Restore the TDAFW Pump to OPERABLE status within 7 days or place the unit in HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 12 hours.

☐ **B** Within 1 hour, initiate action to place the unit in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

☑ **C** Within 2 hours, restore the TDAFW Pump to OPERABLE status or place the unit in HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours.

☐ **D** Restore the TDAFW Pump or the 'A' EDG to OPERABLE status within 72 hours or place the unit in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

Question Misc. Info: MP2*LOIT/LOUT, SRO, 2313A, TS, (CFR-55.43(b)(2), MB-01862, NRC-2002 [K/A 022 CCS, K2.01], NRC-2011, 55.43(b)(2) and (5)

Justification

C: CORRECT, The TDAFW pump being out of service substantially changes the TS required actions for an inop EDG. Instead of TS 3.8.1.1, Action Statement b5 requirements being applied, the requirements of Action Statement b3 must be applied.

A: WRONG; TSAS 3.8.1.1. b3 is more conservative than the actions for an inop TDAFW pump.
Plausible: Chosen if examinees only considered the TS requirements of an inop AFW pump in comparison to the TS requirements of an inop EDG.

B: WRONG; TS 3.0.3 is not applicable because the EDG TS has specific action requirements for the TDAFW pump being inop with an inop EDG.
Plausible: Chosen if examinees think TS 3.0.3 applies due to the increased vulnerability of the plant and lack of TS guidance.

D: WRONG; TSAS 3.8.1.1 b4 cannot be used even though Unit 3 EDGs are considered available.
Plausible: Chosen if examinees believe the alternate requirements of TSAS 3.8.1.1 b4 are now applicable. TSAS 3.8.1.1 b4 allows 14 days to restore the EDG if Unit 3 EDGs are available, or 72 hours if there is a problem with the ability to get power from the Unit 3 EDGs.

References

Tech. Spec. 3.0.5 and 3.6.2.1.

Comments and Question Modification History

07/13/11; Per NRC comments, changed the method by which the "D" CAR Fan is lost to ensure "D" CAR is NOT being tested while "A" D/G is OOS. Reworded the Justifications for choices B and C to ensure there is NO conflict. Removed the reference to "A" EDG in choice C and reworded the question to ensure that only one answer is correct (appropriate).

08/03/11; Per Cliff C. comments, rewrote question to eliminate ambiguity of TS applicability. - rlc

9/14/2011; Reworded stem to make it clear that the requirements of TSAS 3.8.1.1.b.4 were met. This action includes the requirement to verify Unit 3 EDGs and the SBO Diesel are OPERABLE. Changed Terry Turbine to Turbine Driven Auxiliary Feed Pump to use consistent nomenclature. Reworded Choice B to emulate the requirements of TSAS 3.0.3. Reworded choice C to maintain symmetry with choice B. - RJA

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, Standard Appendices, 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, Standard Appendices, then restore Facility 1 CTMT Spray to operation using EOP 2532, LOCA.

Question Misc. Info: MP2*LOIT, LOCA, EOP 2532, RBCCW, NRC-2011, 55.43(b)(5)

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 to be placed in service and preserve the Containment Temperature and Pressure Control Safety Function (the Facility 1 CAR Coolers cannot be restored due to CTMT pressure).

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

Comments and Question Modification History

07/13/11; Per NRC comments, added to the justification for the correct answer (Choice 'D') that the **Facility 1 CAR Coolers cannot be restored due to CTMT pressure.** - rlc

09/19/11; per NRC comments, in choice "C" & "D", added Standard Appendices after EOP 2541. - rlc

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. 1 | ☑ 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 or 2, concurrent with a loss of offsite power and failure of the opposite facility EDG, power from the Turbine Battery will ensure an AFAS can actuate the applicable components to feed the S/Gs.

☑ **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 isolate Main Feed Water flow to the affected S/G.

☐ **C** On a loss of a Vital DC Bus while on Shutdown Cooling, power from the Turbine Battery will ensure that the Flow Control Valves, SI-306 and 657, will remain energized and prevent a loss of Shutdown Cooling.

☐ **D** On a loss of Inverter 1 or 2 concurrent with a Loss Of All AC Power (Station Blackout), power from the Turbine Battery will ensure pressurizer level and RCS Inventory indication is still available.

Question Misc. Info: MP2*LORT, ESD, EOP 2536, 125 VDC, TS Bases, NRC-2011, 55.43(b)(2)

Justification

B - CORRECT; Loss of a Vital DC bus will de-energize Inverters 1 or 2, which are the normal power supplies to VA-10 or VA-20. The Turbine Battery is the back up power supply to VA-10 and VA-20 through Inverters 5 and 6, respectively. Maintaining VA-10 or VA-20 energized will allow MSI to isolate Main Feedwater flow during an ESD with a concurrent loss of offsite power, by automatically closing the applicable Main Feed Reg. Valve.

A - WRONG; This is NOT the bases in Tech Specs.
Plausible: The loss of VA-10 or VA-20 will prevent AFAS from actuating automatically if the opposite Vital AC bus is also lost. 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; Although true, this is NOT the bases in Tech Specs.
Plausible: The loss of a Vital DC Bus will cause a loss of normal power to VA-10 or VA-20 which will result in a loss of Shutdown Cooling. With Inverter 5 and 6 (powered from the Turbine Battery) as the backup power supply, VA-20 and VA-30 will remain energized, preventing a loss of SDC. The examinee should recognize this as a true statement and may believe that this is the basis for the Turbine battery.

D - WRONG; Although true, this is NOT the bases in Tech Specs.
Plausible: The PZR level indication circuit is the only part of the PZR Level Control System that is powered by VA-10 or VA-20 (the majority is powered by non-vital instrument AC). This is to ensure PZR level and RCS inventory indication is not lost during a Station Blackout, when all other sources of vital and non-vital power are lost. The examinee should recognize this as a true statement and may believe that this is the basis for the Turbine battery.

References

TS Bases for 3.8.2.5 (Pg. B 3/4 8-18)

Comments and Question Modification History

07/13/11; Per NRC comments, reworded all 4 choices to eliminate the ability to answer the question based solely on system knowledge. - rlc

09/19/11; per NRC comments, removed "the status of" from choice "D" and added the word "indication". - rlc

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. 1	☑ 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.
Only a log entry of the specific evolution is required.

☐ **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.
Entry into TSAS 3.6.3.1, Containment Isolation Valves, is required.

☐ **C** The PEO sent to operate 2-SI-463 may continue performing other duties, provided the control room is immediately notified when 2-SI-463 has been opened and reclosed and he/she maintains constant communication with the control room.
Only a log entry of the specific evolution is required.

☐ **D** The PEO sent to operate 2-SI-463 may continue performing other duties, provided the control room is immediately notified when 2-SI-463 has been opened and reclosed and he/she maintains constant communication with the control room.
Entry into TSAS 3.6.3.1, Containment Isolation Valves, is required.

Question Misc. Info: MP2*LOIT 2306, CTMT, Integrity, Admin, NRC-2011, 55.43(b)(2) and (5)

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. Additionally, the SRO must make a Narrative Log entry for the existing valve position is required.

B - WRONG; Opening a Containment Isolation valve under the control of a dedicated operator does NOT require entry into the TSAS 3.6.3.1, Note 1: "Containment Isolation Valves may be opened on an intermittent basis under administrative controls."
Plausible; If the examinee does NOT remember the note concerning the administrative control of Containment Isolation Valves, then it would be logical to assume TSAS 3.6.3.1 would apply.

C - WRONG; Operation of 2-SI-463 requires a "Dedicated Operator" (no other duties), due to the administrative impact of a Containment Isolation Valve being open with a subsequent CIAS.
Plausible; There are many Tech. Spec. controlled valves, that do <u>not</u> require a "Dedicated Operator" be present. The examinee may feel that this is one of them.

D - WRONG; Operation of 2-SI-463 requires a "Dedicated Operator" (no other duties), due to the administrative impact of a Containment Isolation Valve being open with a subsequent CIAS. Additionally, opening a Containment Isolation Valve under the control of a dedicated operator does NOT require entry into the TSAS 3.6.3.1.
Plausible; There are many Tech. Spec. controlled valves, that do <u>not</u> require a "Dedicated Operator" be present. The examinee may feel that this is one of them. Additionally, if the examinee does NOT remember the note concerning the administrative control of Containment Isolation Valves, then it would be logical to assume TSAS 3.6.3.1 would apply.

References

OP 2306O, R2C5, Pg. 4, Precaution #2 and Pg. 13, Step 4.3

Comments and Question Modification History

07/13/11; Per NRC comments, reworded to improve distracters. - rlc

09/19/11; per NRC comments, added "and he/she maintains constant communication with the control room" to choices "C" & "D". - 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. 1 | ☑ 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.
0800 - I&C has repaired the problem with the CEDS and reports the dropped CEA is ready for recovery.
0815 - Reactor power is stable at the required level for CEA recovery.
0920 - The dropped CEA has been withdrawn to 162 steps when it stops withdrawing on demand.
0925 - I&C reports approximately 10 minutes is needed to repair a new problem with CEDS and then the misaligned CEA can be fully recovered.

Which of the following actions is the US required to immediately take?

☐ **A** The alternate required actions of the applicable TSAS must be taken. Therefore, insert all other CEAs in Group 7 to 172 steps withdrawn by 0930; otherwise, the unit must be shutdown to MODE 3.

☐ **B** The required actions of the applicable TSAS, once power is reduced, must be taken. Therefore, withdraw the misaligned CEA to at least 170 steps by 1015; otherwise, commence a plant shutdown to MODE 3.

☐ **C** The required actions of the applicable TSAS cannot be met. Therefore, commence a plant shutdown to MODE 3 using boration until the misaligned CEA is withdrawn to at least 170 steps.

☑ **D** The required time limit of the applicable TSAS cannot be met. Therefore, do not withdraw the misaligned CEA and commence a plant shutdown to MODE 3 using boration.

Question Misc. Info: MP2*LOIT, 2556 2 hour time limit on recovery. NRC-2011, 55.43(b)(2) and (5)

Justification

D - CORRECT; TSAS 3.1.3.1 action A.1 is applicable due a misaligned CEA (>10 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. No other CEA motion is allowed while recovering the misaligned CEA.

A - WRONG; AOP 2556 gives specific guidance on the actions to be taken if a misaligned (dropped) CEA is not restored within the required time limit. The AOP states that the reactor must be completely shut down due to the concerns for xenon distortion of power distribution. Therefore, the normal actions taken when a TSAS time limit is going to be missed by only a few minutes is not allowed.
Plausible: The examinee may feel that because the misaligned CEA cannot be **withdrawn** within the time requirements of TSAS 3.1.3.1, the alternative actions required by TSAS 3.1.3.1 must be applicable (i.e., align the Group 7 CEAs within 10 steps of the misaligned CEA within 2 hours of reaching the required power level.)

B - WRONG; TSAS 3.1.3.1 states to reduce power to < 70% within one hour and recover the misaligned CEA within two hours, **not** the <u>next</u> two hours. Therefore, the two hour time clock for TSAS 3.1.3.1 starts as soon as the CEA is misaligned by more than 20 steps. Accordingly, the CEA must be withdrawn to at least 170 steps by 0930.
Plausible: The examinee may believe the two hour time limit to recovery starts once power is reduced to < 70%.

C - WRONG; Commencing a plant shutdown is appropriate; however, unlike other TSAS required actions, if the applicable conditions are corrected shortly after the time limit expires, the reactor must still be completely shut down.
Plausible: The examinee may believe that the usual administrative requirements of not meeting a TSAS can be applied to the CEA misalignment TSAS.

References

AOP 2556, R16C10, Step 1.2 Discussion section and the Caution preceding step 4.28

Comments and Question Modification History

07/13/11; Per NRC comments, reworded stem and choices to improve discriminatory value. - rlc

09/19/11; per NRC comments, modified "(>20 steps)" to "(>10 steps)" in the Justification for the correct answer. - rlc

10/05/11; per validation, modified correct answer to match AOP-2556. - rlc

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 #: 92	Question ID: 1100050	[ ] RO	[x] SRO	[ ] Student Handout?	[x] Lower Order?
	Rev. 0	[x] Selected for Exam		Origin: New	[ ] Past NRC Exam?

While operating in MODE 1, the PORV RC-~~200~~ OPEN annunciator, C-~~11~~ on C-02/3, was suddenly received. The operating crew entered ARP 2590B-043 and observed the following:

- Quench Tank level, pressure, and temperature were normal and stable
- PORV Discharge Temperatures are normal and stable.
- PORV LT/OP Selector Switches are in HIGH.
- Both the Open (red) indication and the Closed (green) indication lights for RC-~~200~~ are lit.

402 B

Posted for examinees during exam. 11/4/11

Which of the following Technical Specification actions must be taken?

[ ] **A** Restore the inoperable channel to OPERABLE status within 30 days or be in HOT STANDBY within the next 12 hours.

[ ] **B** Within 12 hours restore the inoperable PORV Indication to OPERABLE status or close the associated Block Valve.

[ ] **C** Immediately verify RCS leak rate is within Tech. Spec. limits and determine subcooling margin once per 12 hours.

[x] **D** Obtain Quench Tank temperature, pressure, and level, along with PORV discharge temperature, once per shift.

Question Misc. Info: MP2*LOIT, SRO, NNI, PORV, TS, ARP, NRC-2011, 55.43(b)(2) and (5)

Justification

D is CORRECT. ARP 2590B-043 requires the operator to verify Quench Tank parameters are stable. Parameters include temperature, pressure and level. The ARP also directs the SRO to "Reference Technical specification 3.3.3.8 and determine applicability. Tech Spec 3.3.3.8, Accident Monitoring, Table 3.3-11, Action 3, is applicable to the loss of a PORV Position Indicator. The required action is to "obtain Quench Tank level, pressure, and temperature and monitor discharge pipe temperature once per shift to determine valve position."

A is INCORRECT; Action 1 deals with the loss of a Pressurizer Water Level channel.
Plausible: The examinee may remember the first action listed under Accident Monitoring Instrumentation as the appropriate action for a loss of PORV Position Indication. This action may appear to be a reasonable response for a loss of a PORV Position Indication.

B is INCORRECT. The block valve only has to be closed if the PORV is inoperable, not the indication.
Plausible: The examinee may confuse the requirements of the PORV TSAS with the PORV Position Indication TSAS.

C is INCORRECT. Action 2 deals with the loss of an RCS Subcooled/Superheat Monitor and the need to verify RCS leakage out the PORV is not excessive by use of the PPC leak rate program.
Plausible: The examinee may remember the second action listed under Accident Monitoring Instrumentation as the appropriate action for a loss of PORV Position Indication. These actions may appear to be a reasonable response for a loss of a PORV Position Indication.

References

ARP 2590B-043
TS 3.3.3.8

Comments and Question Modification History

07/14/11; Per NRC comments, developed new question to properly align with the K/A statement. - rlc

09/19/11; per NRC comments, removed " and restore the inoperable channel to OPERABLE status within 30 days" from choice "C". Also, to ensure choice symmetry, added "Immediately verify RCS leak rate is within Tech. Spec. limits" to choice "C". - 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: 1100060	☐ RO	☑ SRO	☐ Student Handout?	☑ Lower Order?
	Rev. 0	☑ Selected for Exam		Origin: New	☐ Past NRC Exam?

The following plant conditions exist:

- Power level is 100% and stable.
- All Circulating Water Pumps are in VFD Mode.
- All Circulating Water Pumps are operating at 65% speed.
- Injection temperature is 38°F.
- The date is March 15th.

Suddenly, the "A" Circulating Water Pump trips.

Which of the following is a direct impact of this malfunction and what action(s) must the US direct the crew to perform?

☑ **A** The Unit 2 maximum Circulating Water differential temperature allowed by the NPDES Permit has the potential to be exceeded.
Per AOP 2517, Circulating Water Malfunctions, raise the speed of the remaining Circulating Water Pumps to 100% and crosstie the "A" Waterbox with the "B" Waterbox.

☐ **B** Condenser vacuum will be lost due to the Circulating Water Pump flow limitations imposed during Fish Protection Season.
Per OP 2325, Circulating Water System, raise the remaining Circulating Water Pump speeds to the limit of the NPDES Permit and trip the plant if vacuum exceeds 4.5 inches of Hg absolute.

☐ **C** The lower Circulating Water flow will result in exceeding the maximum discharge temperature allowed by the NPDES Permit.
Per AOP 2517, Circulating Water Malfunctions, raise the speed of ONLY the "B" Circulating Water Pump to 100% and monitor vacuum in each half of the Condenser.

☐ **D** The flow of warm water back to the "A" Service Water Pump will result in exceeding the Ultimate Heat Sink Tech Spec LCO.
Per OP 2325, Circulating Water System, close CW-11H, "A" Waterbox Inlet Isolation, and monitor "A" Circulating Water Bay temperature for Tech Spec entry.

Question Misc. Info: MP2 LOIT 2517, . NRC-2011, 55.43(b)(5)

Justification

A - CORRECT; The normal Unit 2 NPDES Delta-T limit is 32°F; however, when a Circulating Water Pump is off due to a malfunction, the NPDES limit becomes 44°F. When Circulating Water pumps are running at reduced speeds, the unit is intentionally operated at close to the Delta-T limit (30.5-31°F). When a Circulating Water Pump trips, the unit Delta-T will rise. Depending on the conditions, the maximum NPDES Delta-T may be exceeded. In order to maintain less than the max NPDES Delta-T, the remaining Circulating Water pump speeds must be increased. AOP 2517 allows the remaining Circulating Water Pump speeds to be increased to the maximum flow even during Fish Protection Season.

B - WRONG; Condenser vacuum is a concern when a Circulating Water Pump is suddenly lost; however, the given conditions will NOT result in a loss of Condenser vacuum (>4.5 inches Hg absolute)
Plausible: The examinee may think that the flow limitations of the Fish Protection Season apply; however, the NPDES permit allows the remaining Circulating Water Pump speeds to be restored to 100% and NOT the lower limit required by the Fish Protection Program of the NPDES Permit.

C - WRONG; The lower Circulating Water flow <u>will</u> impact the discharge temperature; however, raising the speed of only the "B" Circulating Water Pump will not be enough to prevent exceeding any other NPDES limit because the of the net loss of Circulating Water flow even after raising "B" Circ Water Pump speed.
Plausible; The examinee may believe that Circulating Water flow must be maintained at <u>some</u> lower flow rate per the NPDES Permit. Raising the flow through the "A" Condenser <u>will</u> result in a more balanced vacuum on both halves of the Condenser.

D - WRONG; Warm water <u>will</u> flow back to the "A" Service Water Pump; however, it will mix with the relatively cold bay water and will NOT challenge the basis for the Ultimate Heat Sink temperature limit (75°F) with the given conditions.
Plausible; OP 2325, Circulating Water System, states that securing a Circulating Water Pump will cause warm water to flow back to the associated bay and may have an impact on the Ultimate Heat sink temperature. The procedure also states that closing the affected Waterbox Inlet valve will eliminate the problem. However, AOP 2517, Circ. Water Malfunctions, does not discuss this concern.

References

AOP 2517, Circulating Water Malfunctions, Section 5.0
OP 2325A, Circulating Water System

Comments and Question Modification History

8/3/11; Per NRC comments, developed new question to improve SRO descriminatory value. - RJA

Question #: 93 **Question ID: 1100060** ☐ RO ☑ SRO ☐ Student Handout? ☑ Lower Order?

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

09/19/11; Per NRC comments, replaced "may" with "will" in choices "B", "C" & "D". Fixed typo in Justification for choice "C", changing "C" circ. pump to "B". Slight modification of wording in 2nd sentence of choice "A" to enhance readability. Added to Justification of choice "D" to increase understanding of the reason for this choice being incorrect and why it is plausible. - rlc

09/28/11; per NRC comments, removed the word "likely" from the Justification for choice "C". - rlc

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

The plant is currently at 28% power, starting up after a forced outage.

Are the Main Feed Regulating Valve Bypass Valves required to be Open or Closed for these conditions and why?

☐ **A** CLOSED. Due to the potential for an Excess Steam Demand Event challenging the Reactivity Control Safety Function.

☑ **B** CLOSED. To ensure an Excess Steam Demand Event Inside CTMT does NOT challenge the CTMT Temperature and Pressure Safety Function.

☐ **C** OPEN. Due to the instability of automatic feedwater control at low steam demands.

☐ **D** OPEN. To ensure automatic transition to the Main Feed Regulating Valves at low power.

Question Misc. Info: MP2*LOIT* 2204, 55.43(b)(1)

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. The FSAR reference states that the analysis only took into account a MFRV Bypass being opened at or below 25% power

A - WRONG; This part of the accident analysis is concerned with an ESD inside containment, NOT just an ESD. Therefore the CTMT barrier is the overriding concern, NOT reactor restart from an excessive RCS cooldown.
Plausible; The correct answer does involve the impact on the accident analysis of a Steam Line Break.

C - WRONG; Must be closed > 25% power. As long as power level is changed within the rate specified by procedure, the Feed Control System should maintain stable S/G levels.
Plausible; Feedwater control at low power levels is inherently unstable due to the effects of increased shrink and swell and for many years could not be placed in automatic control until about 15% power.

D - WRONG; Both FRV Bypass valves must be CLOSED above 25% power.
Plausible; The examinee might believe the FRVs can't transition at low power levels.

Correction posted for examinees during exam
RP 11/4/11

References

OP 2204, R23C5, Precaution 3.6 and Reference 6.9.
FSAR, R25.1, Containment Analysis - Steam Line Break Analysis, Chap. 14.8.2.1.6.2, Item #3

Comments and Question Modification History

07/18/11; Per NRR comments, rewrote stem and Choices "A" & "B" (with minor revisions to "C" & "D") to minimize the ability to answer the question based solely on knowledge of the procedural preacaution. - rlc

9/19/2011; Per NRC comments, reworded question stem to solicit whether the Main FRV Bypass Valves aer required to open or clsoed and why. Reworded choices to provide a choices of either OPEN or CLOSED. Changed the reason in Choice D to be more plausible. - rlc

09/28/11; per NRC comments, added "Reference 6.9" to the given question references and changed the 10CFR55.43(b) alignment from 55.43(b)(**5**) to 55.43(b)(**1**). - 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 2.1.32 **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: 56637	☐ RO ☑ SRO	☐ Student Handout?	☑ Lower Order?
	Rev. 1	☑ Selected for Exam	Origin: Bank	☐ Past NRC Exam?

The following plant conditions exist:

- A refueling outage is in progress.
- Core off-load is near completion.
- The Upender is moving to the vertical position on the Refuel side.
- A fuel assembly is on the hoist and has just been withdrawn from the core.
- Refuel pool level suddenly begins lowering rapidly.
- The Control Room reports that a S/G Nozzle Dam has failed.

In accordance with AOP 2578, Loss of Refuel Pool and Spent Fuel Pool Level, which of the following actions is the highest priority for placing the Fuel Bundle in a safe location?

☐ **A** Continue with the step to place the Fuel Bundle in the Upender and send it to the Spent Fuel Pool.

☐ **B** Move the Refuel Machine to any open location in the South Saddle and fully lower the fuel bundle.

☑ **C** Select any open location in the vessel and fully lower the fuel bundle.

☐ **D** Move the refuel machine to the "Safe Point" and fully lower the fuel bundle.

Question Misc. Info: MP2*LOIT, 2303, 2578, FH, Refuel, NRC-2011, 55.43(b)(4), (5), and (7)

Justification

C is CORRECT. AOP 2578 provides the SRO with choices as to where the Fuel bundle should be placed in the event Refuel level is lowering. The note prior to the step on deciding where to place the fuel bundle, provides a priority for placing the fuel bundle in the reactor vessel, given the present conditions.

A is incorrect. In order of priority, the second choice states, "If near the Upender <u>and</u> the transfer carriage is in Containment <u>and</u> time allows, position the fuel assembly in the Upender and transfer to the Spent Fuel Pool. The note following this step states that it takes up to 50 minutes to close the Transfer Tube Isolation Valve, RW-280. Because of the fuel bundle location (NOT near the Upender) and the time involved in sending the bundle to the SFP, this option is impractical.
Plausible: The examinee may feel that moving the bundle to the SFP is more conservative because it will be removed from the RCS where the loss of level is occurring. The Transfer Tube Isolation, RW-280 will be closed whether the bundle is transferred to the SFP or it stays in CTMT.

B is incorrect. The last option listed in AOP 2578 states, "With the Refuel machine in the manual mode, move to a clear area in the South Saddle and lower the assembly." The Refuel Machine is NOT in Manual at this time. Additionally, it would take longer to get to the South Saddle than it would to get to any open location in the vessel.
Plausible: The Refuel Machine may be placed in the manual mode simply with the push of a button. The examinee may feel that, with the loss of inventory from the RCS, any open location in the South Saddle may be a better storage location.

D is incorrect. The "Safe Point" is a specific location programmed into the Refuel Machine computer and is the third option listed in AOP 2578. The procedure also states that this option is viable if unable to place the assembly in the vessel or transfer it to the SFP.
Plausible: The examinee may not remember specifically where the "Safe Point" is. As a result, he/she may feel that the "Safe Point", due to it's name, is the appropriate designated location for a Fuel Assembly in an emergency.

References

AOP 2578

Comments and Question Modification History

07/15/11, Replaced question as a result of NRC comment. Original question was not an exact K/A match. RJA

09/19/11; per NRC comments, modified stem question statement to ask for the "highest prioirty" for placing the fuel bundle in a safe location.

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. 2	☑ 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 been isolated due to a header rupture in the Enclosure Building.
- 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.
It has been 40 minutes since the trip from 100% power.

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** Tcold can NOT be lowered more than approximately 40°F over the next 20 minutes from this point, once the RCS cooldown is reinitiated.

☐ **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.

Question Misc. Info: MP2*LOUT, 2541, Appnd. 17, TS Bases, TRM, NRC-2003 [K/A E09 2.1.33], NRC-2011, 55.43(b)(2) and (5)

Justification

D - CORRECT; The shutdown Margin curves take into account the most reactive CEA has stuck out, but ONLY <u>one</u> 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; IAW EOP 2534, SGTR, the NOTE prior to step 26 (the step to reinitiate the cooldown) the temperature used for the Cooldown Limit is "reset" to the existing Tcold when Thot was reduced to less than 515 °F. Therefore, a plant cooldown should continue from 485°F, not to exceed the TS limit from that point on.
Plausible; This would be the required action under non-accident conditions [100% power Tcold = 545°F; 545°F - 100°F limit for one hour = 445°F limit for the hour; Tcold is now at 485°F; therefore, RCS can cooldown 40 more degrees in the next 20 minutes left and meet the one hour TS Limit].

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

07/14/11; Per NRC comments, added "**It has been 40 minutes since the trip from 100% power.**" to the stem and reworded **Choice "A"**. - rlc

09/19/11; per NRC comments, reworded the Justification for the reason choice "A" is incorrect to include the reference that describes when the cooldown rate is reset.

NRC K/A System/E/A **System** 2.2 Equipment Control

Generic K/A Selected

NRC K/A Generic **System** 2.2 Equipment Control

Question #: **96**	**Question ID: 3100015**	☐ **RO**	☑ **SRO**	☐ **Student Handout?**	☐ **Lower Order?**
	Rev. 2	☑ **Selected for Exam**		**Origin: Bank**	☐ **Past NRC Exam?**

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: 1100052	☐ RO	☑ SRO	☐ Student Handout?	☐ Lower Order?
	Rev. 1	☑ Selected for Exam		Origin: New	☐ Past NRC Exam?

During normal, MODE 1 operation, the following indications are observed:

- Both Containment High Range Radiation Monitors (RM-8240/8241) on C101 have red lights energized.
- POST INCIDENT RAD. MONITOR HI/FAILURE, annunciator on C-02 is in alarm.
- All other containment radiation monitors indicate normal levels.

What action must be taken for this condition?

☐ **A** Per TSAS 3.4.8, Reactor Coolant System Specific Activity, Within one hour, verify the specific activity of the primary coolant is </= 1.0 µCi/gram DOSE EQUIVALENT I-131.

☐ **B** Per TSAS 3.3.3.1, Radiation Monitoring, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

☑ **C** Per TSAS 3.3.3.1, Radiation Monitoring, initiate the preplanned alternate monitoring method within 72 hours and restore the inoperable channel to OPERABLE status within 7 days.

☐ **D** Per TSAS 3.0.3, within 1 hour, initiate ACTION to commence a plant shutdown and subsequent cooldown. Be in at least COLD SHUTDOWN with the subsequent 32 hours.

Question Misc. Info: MP2*LOIT Rad. Mon., SRO, NRC-2005 [K/A 061, ARM, AA2.01], NRC-2011, 55.43(b)(4)

Justification

C is correct. With both red and yellow lights energized, the radiation monitors indicate a common failure and are, therefore, inoperable. TSAS 3.3.3.1, Table 3.3-6, ACTION 17 applies. The alternate method of monitoring EPI-FAP11, Core Damage Assessment specifies using the Main Steam Line Radiation Monitors as an alternate monitoring method.

A is incorrect. The requirements of this TSAS ACTION do NOT apply for just this indication.
Plausible: The examinee may feel that the Containment High Range detectors are indicating a possible high radiatino in the CTMT due to a severe crud burst.

B is incorrect. This action is only required for an actual high radiation with the Control Room Area rad monitor inoperable.
Plausible: The examinee may confuse the ACTION associated with the Containment High Range Rad Monitors with the ACTION of the Control Room Radiation Monitors, which is listed in the same table.

D is incorrect. TSAS 3.0.3 does NOT apply. Even though there are 2 Containment High Range Radiation Monitors and only 1 is required to OPERABLE, the TS Action is for <u>1 less than the minimum required</u>, which is <u>none</u> OPERABLE.
Plausible: The examinee may believe that, because there are only 2 Containment Radiation Monitors, then both must be OPERABLE. If only 1 is OPERABLE, then Tech Spec LCO 3.3.3.1 must apply. If <u>both</u> are inoperable, then TSAS 3.0.3 must apply.

References

1. OP 2383B, Area Radiation Monitors

Comments and Question Modification History

07/18/11, Per NRC comment (system knowledge only), replaced question. RJA

09/19/11; per NRC comments, modified stem to more closely match the K/A statement. - rlc

09/28/11; per NRC comments, reworded stem and choices "A" & "B" for possibility that rad monitors are displaying an actual high radiation. - 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 #: 100	Question ID: 76423	☐ RO	☑ SRO	☐ Student Handout?	☑ Lower Order?
	Rev. 3	☑ 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 yet evaluated constraining conditions.

☐ **C** Personnel Accountability is NOT yet completed.

☐ **D** The Operational Support Center has NOT yet been activated.

Question Misc. Info: MP2*LORT*EPlan Security Event, NRC-2011, 55.43(b)(5)

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

09/19/11; per NRC comments, reference used is ***NOT*** Safeguards Information and, as such, question does ***NOT*** have to labeled as "SENSITIVE INFORMATION". Also, modified correct answer to be more in line with the protective action decisions given in the reference.

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.