Question #: **1** **Question ID: 2018017** **Rev. 0** ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/14/2018 8:44:54 AM ☑ **RO** ☐ **SRO** **Origin: New** ☐ Past NRC Exam?

A mid-cycle reactor startup is in progress.
The Main Turbine trip has not been reset yet due to EHC control issues.

Then, TCB-9 trips open (due to an internal breaker failure).

The Shift Manager directs the reactor startup be aborted and all CEAs be simultaneously inserted, until the existing plant problems are rectified.
The RO then presses all four reactor trip push buttons and all CEAs fully insert.

Which of the following describes the plant status at this time?

☐ **A** All of the CEDS ACTMs have completely de-energized.

☐ **B** The steam supply to the MSRs, MS-2A/B valves go closed.

☐ **C** The Main Feed Reg. Bypass Valves will ramp to 40% open.

☑ **D** The two CEDS MG sets are no longer running in parallel.

Question Misc. Info: MP2*ILT [ENTER Searchable Info]

Justification

A - WRONG; If both MG sets are still connected to the CEA busses, then the "Backup" ACTM power supply is still energized, maintaining power to the ACTMs.
Plausible; The student may believe that with the loss of the normally closed TCB-9, the ACTM backup power supply would be lost on a plant trip. The backup power supply is actually hard wired through the TCB-10 breaker cubical and is therefore unaffected when TCBs 1 - 8 open.

B - WRONG; MS-2A/B steam supply to the MSRs are interlocked closed until the Main Turbine load is above 10%. Therefore they would not have been open at this time.
Plausible; The student may confuse the specific valves that are opened to warm up the steam plant once the plant reaches NOP/NOT. On a normal plant trip, it is important to ensure these valves go closed or an uncontrolled RCS cooldown could occur.

C - WRONG; The Main Feed Reg. Bypass Valves only ramp to 40% on a Main Turbine Trip signal. If the turbine trip hasn't been reset, a reactor trip cannot trigger this signal.
Plausible; The student may remember the valves ramp to 40% on a plant trip, but not the specific signal that drives them there.

D - CORRECT; TCBs 1 - 8 are normally opened on a reactor trip (pressing the four trip buttons). Normally, both MG sets will remain in parallel to allow reclosing of TCBs 1 - 8 post-trip. However, with TCB-9 open, the MG sets are no longer in parallel as their busses are no longer cross-tied once the other 8 TCBs open.

References

ARP 2590C-075_R0C0, OP 2302A, R20C0, St. 4.6.1b and 4.6.3

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 007 000007 (EPE 7; BW E02&E10; CE E02) Reactor Trip, Stabilization, Recovery / 1

Number EK2.02 **RO** 2.6 **SRO** 2.8 **CFR Link** (CFR 41.7 / 45.7)

EK2.02 Knowledge of the interrelations between a reactor trip and the following: Breakers, relays and disconnects

Question #: **2** **Question ID: 53793** **Rev. 1** ☐ Student Handout? ☐ Lower Order?

Last Edited: 5/31/2018 12:43:39 PM ☑ **RO** ☐ **SRO** **Origin: Mod** ☐ Past NRC Exam?

The plant tripped due to a Small-Break LOCA on the top of the Pressurizer.
On the trip, 24C and 24D did not transfer to the RSST and are now powered by their Emergency Diesel Generators.

The following conditions now exist:

- RCS Pressure is 900 psia and stable.
- RCS temperature is 532°F and stable.
- Pressurizer level is 85% and rising.
- Reactor vessel level 69% and lowering.
- Containment pressure is 5 psig and rising slowly.
- All other plant systems and components are functioning as designed.

Which one of the following describes the RCP trip strategy, and the reason for it, under the existing conditions?

☐ **A** Only two RCPs must be secured.
Lower the amount of water mass inventory lost through the break with minimal safety injection flow.

☑ **B** All four RCPs must be secured.
Minimize potential for long term RCP loss due to damage from excessive operation with cavitation.

☐ **C** All four RCPs must be secured.
Allow for greater RCS de-pressurization to lower the RCS leak rate and raise Safety Injection flow.

☐ **D** Only two RCPs must be secured.
Lower the amount of heat being added to the RCS that must then be removed by safety injection.

Question Misc. Info: MP2 LOIT E32-01-C, MB-5941

Justification

A - WRONG; RCS pressure and temperature indicate subcooling is zero, which means the RCP NPSH requirements can NOT possibly be met. Therefore all RCPs must be secured.
Plausible; The student may recall that only 2 RCPs would be secured to ensure water is NOT 'pumped' out the break, therefore limiting the amount of inventory lost during a LOCA. If the SB-LOCA had occurred anywhere else in the system, RCS pressure would probably still be above minimum NPSH and this would be the correct action and the reason for it.

B - CORRECT; The existing RCS temperature and pressure indicate saturated conditions. Based on this, the RCS conditions do NOT meet minimum NPSH requirements for continued RCP operation and all four pumps must be secured. Excessive RCP operation below minimum NPSH requirements could lead to RCP damage due to cavitation, making the pumps unavailable long term when plant conditions improve.

C - WRONG; A hole in the top of the PZR will cause an almost immediate loss of normal RCS pressure control using PZR Heaters or Spray Flow. Therefore, the only real means to lower RCS pressure faster than inventory loss would be by RCS Cooldown. However, this would improve RCS conditions toward the values that allow for continued RCP operation.
Plausible; The student may recognize that HPSI flow will may not be enough to prevent core uncovery. Therefore, this is a logical reason for securing RCPs with a Small Break LOCA in any other location in the RCS that maintains a bubble in the Pressurizer.

D - WRONG; Two RCPs cannot remain operating with RCS conditions violating minimum NPSH requirements.
Plausible; The student may see this as a correct reason for securing RCP operation if secondary heat removal is challenged. Based on the loss of condenser vacuum (MSI), secondary heat removal capacity would be limited to the amount of inventory in the Condensate Storage Tank (CST). Limiting RCS heat input would increase the time available for CST makeup.

References

EOP 2541, Figures 1 and 2.
OP 2260, R15C0, RCP Trip Strategy

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 008 000008 (APE 8) Pressurizer Vapor Space Accident / 3

Number AK3.04 **RO** 4.2 **SRO** 4.6 **CFR Link** (CFR 41.5,41.10 / 45.6 / 45.13)

AK3.04 Knowledge of the reasons for the following responses as they apply to the Pressurizer Vapor Space Accident: RCP tripping requirements

Question #: **2**	***Question ID:*** ***53793***	**Rev.** **1**		☐ **Student Handout?**	☐ **Lower Order?**	
Last Edited: 5/31/2018 12:43:39 PM		☑ ***RO***	☐ ***SRO***	***Origin:*** ***Mod***	☐ **Past NRC Exam?**	

Question #: **N/A** | **Question ID: 53793** | **Rev. 0** | ☐ Student Handout? | ☐ Lower Order?

☑ **RO** ☐ **SRO** | **Origin: Parent** | ☐ Past NRC Exam?

PARENT

Given the following conditions:
* A Loss of Coolant Accident has occurred.
* RCS Pressure is 1500 psia.
* Two (2) Reactor Coolant Pumps (RCPs) have been secured.

Which one of the following describes why the RCPs are secured under these conditions?

☑ **A** Lower the amount of water mass inventory lost through the break, therefore enhancing efforts to keep the core covered.

☐ **B** Raise the flow of steam (instead of two-phase mixture) from the break, therefore enhancing heat removal from the core.

☐ **C** Lower the cold leg pressure head, therefore, enhancing safety injection system performance at higher flow rates.

☐ **D** Lower the amount of heat being added to the RCS that must then be removed by safety injection.

Question Misc. Info: MP2 LOUT E32-01-C, MB-5941

Justification

A; Correct - RCPs are secured to ensure water is NOT 'pumped' out the break (worst case is bottom of hot leg), therefore limiting the amount of inventory lost during a LOCA. B; Wrong - more energy would be removed in a two-phase mixture. C; Wrong - injection flow sees "RCS-to-RWST Head" delta-p, not loop delta-p generated by operating RCPs. D; Wrong - correct reason for normal plant cooldown operations.

Question References not yet listed.

NO Comments or Question Modification History at this time.

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

Number EK3.11 **RO** 3.3? **SRO** 3.4? **CFR Link** (CFR 41.5 / 41.10 / 45.6 / 45.13)

Knowledge of the reasons for the following responses as the apply to the Large Break LOCA: NC and PC

Question #: **3** **Question ID: 2014003** **Rev.** 1 ☐ **Student Handout?** ☐ **Lower Order?**

Last Edited: 6/25/2018 1:31:52 PM ☑ **RO** ☐ **SRO** **Origin: Mod** ☐ **Past NRC Exam?**

The plant has experienced a Small Break Loss of Coolant Accident (SBLOCA). The core is being cooled by two phase natural circulation flow (reflux boiling). The Steam generator is at 500 psia.

Which of the following would raise the Natural Circulation cooling rate?

☐ **A** Raising HPSI injection flow.

☐ **B** Lowering Steam Generator level.

☑ **C** Lowering Steam Generator pressure.

☐ **D** Raising Pressurizer pressure.

Question Misc. Info: MP2*LOIT Bank Palo Verde, NRC-2014

Justification

A - INCORRECT: Raising HPSI injection flow will not raise the natural circulation cooling rate. Changing RCS parameters will have negligible effect on reflux boiling .
PLAUSIBLE: Because raising HPSI injection flow would raise cooling to the RCS but it would not raise the natural circulation cooling rate. This is arguably correct because we monitor RCS conditions to determine the rate of core cooling and raising injection flow would indeed raise RCS/core cooling.

B - INCORRECT: Although the SG is a saturated system like the pressurizer, lowering Steam Generator level will not lower SG pressure as it does when lowering pressurizer level. Therefore, this action will not raise the natural circulation cooling rate.
PLAUSIBLE: Because the examinee may understand that lowering Steam Generator pressure will raise feedwater flow and therefore will raise the natural circulation cooldown rate. But the steam generator is already at a pressure to allow feeding with the condensate pumps without lowering steam generator pressure lower. The student must know the discharge pressure of the condensate pumps.

C - CORRECT: Lowering Steam Generator pressure will raise the rate of heat removal from the RCS, which will raise the natural circulation cooling rate. Reflux boiling is the process of steam going up the SG tubes, condensing and falling back into the RCS. The greater the heat removal rate from the Steam Generators, the greater the NC cooling rate.

D - INCORRECT: Raising Pressurizer pressure will not raise the natural circulation cooling rate. Changing RCS parameters will have negligible effect on reflux boiling.
PLAUSIBLE: The student may believe that eliminating the "2-phase" condition of the RCS would raise RCS cooling. Although this is an eventual goal, initially it would result in lower safety injection flow, which would lower core cooling.

References

EOP 2532 R34, Loss of Coolant Accident

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 009 000009 (EPE 9) Small Break LOCA / 3

Number EK1.01 **RO** 4.2 **SRO** 4.7 **CFR Link** (CFR 41.8 / 41.10 / 45.3)

EK1.01 Knowledge of the operational implications of the following concepts as they apply to the small break LOCA: Natural circulation and cooling, including reflux boiling

Question #: **N/A** **Question ID: 2014003** **Rev. 0** ☐ Student Handout? ☐ Lower Order?

☑ **RO** ☐ **SRO** **Origin: Parent** ☑ Past NRC Exam?

PARENT

The Unit has transitioned to two phase natural circulation flow (reflux boiling) due to a small break LOCA with inadequate HPSI flow.

The crew can enhance reflux boiling heat removal by increasing…

☐ **A** RCS T-cold to > 550°F.

☐ **B** PZR level from 15 to 55%.

☑ **C** SG NR level from 10% to 50%.

☐ **D** PZR pressure from 1500 to 1600 psia.

Question Misc. Info: MP2*LOIT Bank Palo Verde, NRC-2014

Justification

A - Incorrect: Changing RCS parameters will have negligible effect on reflux boiling.

B - Incorrect: Changing RCS parameters will have negligible effect on reflux boiling.

C - CORRECT: Reflux boiling is the process of steam going up the SG tubes, condensing and falling back into the RCS. The greater the tube coverage the greater the cooling.

D - Incorrect: Changing RCS parameters will have negligible effect on reflux boiling.

Question References not yet listed.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 009 000009 (EPE 9) Small Break LOCA / 3

Number EK1.01 **RO** 4.2 **SRO** 4.7 **CFR Link** (CFR 41.8 / 41.10 / 45.3)

EK1.01 Knowledge of the operational implications of the following concepts as they apply to the small break LOCA: Natural circulation and cooling, including reflux boiling

Question #: **4** **Question ID: 2018001** Rev. 0 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/19/2018 1:06:13 PM ☑ **RO** ☐ **SRO** **Origin: New** ☐ Past NRC Exam?

The plant has experienced a Design Bases Large Break Loss of Coolant Accident (LBLOCA). The Loss of Coolant Accident procedure has been entered and all actions associated with the Sump Recirculation Actuation Signal (SRAS) were completed.

Which of following is/are the minimum pump(s) required for core cooling.

☑ **A** One HPSI pump

☐ **B** One HPSI pump and One LPSI pump

☐ **C** Two HPSI pumps

☐ **D** One HPSI pump and one Charging pump

Question Misc. Info: New question for the 2018 NRC written examination.

Justification

A - CORRECT: One HPSI pump is required for core cooling. In SRAS the LPSI pumps are shut off and are no longer needed for core heat removal. The assumptions for a design bases LOCA is that offsite power is lost and only one EDG is available to power a vital 4.16 KV bus (E32-01-C, page 5). Therefore one HPSI pump is not available. The swing HPSI pump may or may not have power depending on were it is aligned but either way it does not automatically start and is not required. EOP 2541, Appendix 2, Figures, Figure 5 "Minimum ECCS Flow Requirements for Decay Heat Removal" shows that after 40 minutes post trip the safety injection flow required for core heat removal is 300 gpm. One HPSI pump is capable of providing 600 gpm flow at 870 psig (ECC-01-C).

B - INCORRECT: One HPSI pump and One LPSI pump is not correct because while the HPSI pump is required for RCS heat removal the LPSI pump isn't. The LPSI pumps are stopped by the SRAS signal and the LOCA procedure ensures the LPSI pumps are off once the SRAS signal is actuated.
PLAUSIBLE: The examinee could think that a LPSI pump is needed because it is a LBLOCA. The examinee could also understand that LPSI pumps are stopped in SRAS but also understand that a LPSI pump is also started post SRAS for boron precipitation. They could easily think that boron precipitation control is required for core cooling but it isn't. Boron precipitation by the LPSI pumps returns the crystallized boron back into the RCS water by flushing water down into the core from the hot leg. If LPSI pumps are not available for boron precipitation one of the HPSI pumps can be used.

C - INCORRECT: Two HPSI pumps is not correct since only one is required because core heat removal requirements are within the capacity of a single HPSI pump. EOP 2541, Appendix 2, Figures, Figure 5 "Minimum ECCS Flow Requirements for Decay Heat Removal" shows that after 40 minutes post trip the safety injection flow required for core heat removal is 300 gpm.
PLAUSIBLE: The examinee could think that two facilities of Safety Injection are required. Especially since in a normal, event without the loss of a vital bus, both the HPSI pump would be operating.

D - INCORRECT: One HPSI pump and one Charging pump is not correct since only one HPSI pump is required. Core heat removal requirements are within the capacity of a single HPSI pump. A charging pump is not required in a Design Bases LOCA for core heat removal. Charging pumps were originally classified as an ECCS subsystem, but over time, the flow from the pumps was removed from the safety analysis. Therefore, flow from the charging pumps is no longer required for design basis accident mitigation. The loss of coolant accident analysis has been revised and no credit is taken for charging pump flow. As a result, the charging pumps no longer meet the first three criteria of 10CFR 50.36 (c)(2)(ii) as design basis accident mitigation equipment required to be controlled by Technical Specifications.
PLAUSIBLE: The examinee could understand that only one facility of ECCS is required and this includes a charging pump but does not included a LPSI pump post SRAS. The examinee could think that a Charging pump is part of the ECCS since it was at one time and not so distance in the past.

References

ECC-01-C Lesson Text R5

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 011 000011 (EPE 11) Large Break LOCA / 3

Number EK2.02 **RO** 2.6* **SRO** 2.7* **CFR Link** (CFR 41.7 / 45.7)

EK2.02 Knowledge of the interrelations between the and the following: Large Break LOCA: Pumps

Question #: **5** ***Question ID:*** ***1685734*** **Rev.** **0** ☐ **Student Handout?** ☐ **Lower Order?**

Last Edited: 6/14/2018 8:45:54 AM ☑ ***RO*** ☐ ***SRO*** ***Origin:*** ***Bank*** ☑ **Past NRC Exam?**

Millstone Unit 2 is operating at 100% power, when the following indications are observed on Main Control Panel C-02/3:

- "A" Reactor Coolant Pump current meter goes full-scale.
- "A" Reactor Coolant Pump Breaker automatically trips open.
- The Reactor automatically trips.

Which of the following describes the basis for the reactor trip at this time?

☐ **A** RCS pressure will increase, challenging RCS pressure design limits.

☐ **B** RCS Thot will increase, causing ASI to shift and challenge peak centerline temperature limits.

☐ **C** RCS loop flow distribution will be asymmetric, challenging RPS Thot and Tcold input error limits.

☑ **D** DNBR will be lower, challenging fuel temperature design limits.

Question Misc. Info: MP2*LOIT*T RCS initial response to locked rotor, NRC-2016

Justification

A - WRONG: RCS design pressure limits are bounded by a load reject without a reactor trip scenario.
PLAUSIBLE: Examinee may focus on the sudden RCS pressure rise, due to the RCS heatup, potentially challenging the High Pressure trip setpoint.

B - WRONG: The LPD trip is designed to protect against PCT challenges due to a high ASI.
PLAUSIBLE: Examinee may believe the shift in ASI, caused by the expected rise in Thot, will challenge the PCT limit.

C - WRONG: The loss of input conservatism in one loop will be offset by the over conservative input in the other loops.
PLAUSIBLE; Examinee may recognize that the loss of flow in one loop will degrade that loop's input to the LSSS.

D - CORRECT; Loss of flow causes the DNBR to DECREASE (closer to DNB) due to the rise in RCS temperature. DNBR minimum is one of the factors assumed in the calculated maximum fuel clad temperature.

References

Tech Spec Bases for RCS Low Flow Trip.;
LP CPD-00-C, R6C0, Pg 29 and 30.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 015 000015 (APE 15) Reactor Coolant Pump Malfunctions / 4

Number AK1.03 **RO** 3.0 **SRO** 4.0* **CFR Link** (CFR 41.8 / 41.10 / 45.3)

AK1.03 Knowledge of the operational implications of the following concepts as they apply to Reactor Coolant Pump Malfunctions (Loss of RC Flow): The basis for operating at a reduced power level when one RCP is out of service

Question #: **6** **Question ID:** **78836** **Rev.** 2 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/14/2018 8:46:13 AM ☑ **RO** ☐ **SRO** **Origin:** **Mod** ☐ Past NRC Exam?

The plant is at 100% power, normal operation, when annunciators C02/3-C15, "LETDOWN FLOW LO", and C02/3-C8, "LETDOWN PRES HI/LO" alarm.
The crew suspects the Letdown Backpressure transmitter has failed LOW. Based on the suspected cause, the US directs Charging and Letdown be secured.

Prior to <u>any</u> operator action being taken (CVCS is NOT yet secured), which of the following system responses would result if the suspected instrument failure was indeed the cause of the alarm?

☑ **A** Letdown relief valve would open, VCT level would begin to slowly lower, Pressurizer level would remain constant.

☐ **B** Letdown flow would isolate, VCT level would begin to slowly lower, Pressurizer level would begin to slowly rise.

☐ **C** Letdown relief valve would open, Indicated letdown flow remains constant, Pressurizer level would remain constant.

☐ **D** Letdown flow indicates zero, actual flow is reduced to limiter minimum, Pressurizer level would begin to slowly rise.

Question Misc. Info: MP2*LORT 2304, ARP, Letdown, Backpressure Control

Justification

A - CORRECT; The backpressure controller would completely close the backpressure control valves, dropping letdown flow <u>seen by the flow transmitter</u> to zero. However, actual letdown flow would remain constant because the letdown relief valve, located between the backpressure control valves and the letdown flow control valves, would lift - diverting what ever letdown flow is allowed past the flow control valves to rad waste. With letdown flow bypassing the VCT but remaining constant, VCT level would slowly lower while pressurizer level would remain constant.

B - WRONG; PZR level remains constant because the letdown relief is capable of relieving maximum letdown flow, if required. Therefore, actual letdown flow would not change but the flow is not going to the VCT, causing VCT level to drop.
PLAUSIBLE; This would be the effect if the student believed letdown flow was actually going down.

C - WRONG; Letdown flow is sensed downstream of the letdown relief valve and would, therefore, see none of the flow passing through the relief.
PLAUSIBLE; The failed instrument has nothing to do with letdown flow indication. It is only the effect of this failure that causes letdown flow indication to go to zero. The Student may think indicated" letdown flow is sensed upstream of the relief valve and recognize that letdown flow has not really changed and therefore believe flow indication shows this.

D - WRONG; The backpressure controller has "integral action" or auto reset. The controller will continue to close the backpressure control valves until pressure returned to setpoint (~ 350 psig) or the valves go fully closed.
PLAUSIBLE; The backpressure controller is one of only two system controllers that has "integral action" programmed in. If the student believes the back pressure control circuit is designed similar to the letdown flow controller (to prevent lifting of the letdown relief valve), this would be correct.

References

ARP 2590B-059_R1C0 (C02/3-C15).
CVCS Training Drawing

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 022 000022 (APE 22) Loss of Reactor Coolant Makeup / 2

Generic K/A Selected

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

Number 2.4.46 **RO** 4.2 **SRO** 4.2 **CFR Link** (CFR: 41.10 / 43.5 / 45.3 / 45.12)

G2.4.46 Ability to verify that the alarms are consistent with the plant conditions.

Question #: **N/A** **Question ID:** **78836** Rev. 1 ☐ Student Handout? ☐ Lower Order?
☑ **RO** ☐ **SRO** **Origin:** **Parent** ☐ Past NRC Exam?

PARENT

The plant is at 100% power, normal operation, when the Letdown Backpressure Controller transmitter fails to 200 psig.|||Which one of the following would be the expected system response to this instrument failure?|

☑ **A** Indicated letdown flow would go to zero (0) and Pressurizer level would remain constant.

☐ **B** Indicated letdown backpressure would rise to maximum and Pressurizer level would slowly rise.

☐ **C** Indicated letdown backpressure would lower to 200 psig and letdown flow would go to the limiter minimum.

☐ **D** Indicated letdown flow would remain constant and letdown backpressure would rise to maximum.

Question Misc. Info: MP2*LORT 2304, ARP, Letdown, Backpressure Control

Justification

A - CORRECT; The backpressure controller would completely close the backpressure control valves, dropping indicated letdown flow to zero. However, the letdown relief valve, located between the backpressure control valves and the letdown flow control valves, would lift - diverting what ever letdown flow is allowed past the flow control valves to rad waste. Therefore, pressurizer level would remain constant.

B - WRONG; PZR level remains constant because the letdown relief is capable of relieving maximum letdown flow, if required. Therefore, initial letdown flow is irrelevant and would not change.

C - WRONG; The backpressure controller has "integral action" or auto reset. The controller will continue to close the backpressure control valves until pressure returned to setpoint (~ 350 psig) or the valves go fully closed.

D - WRONG; Letdown flow is sensed downstream of the letdown relief valve and would, therefore, see none of the flow passing through the relief. Also, the only indication of letdown backpressure is the failed instrument.

Question References not yet listed.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System**

Number	RO	SRO	CFR Link

Question #: **7** **Question ID: 2018002** **Rev. 0** ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/20/2018 10:30:11 PM ☑ **RO** ☐ **SRO** **Origin: New** ☐ Past NRC Exam?

The plant is currently on Shutdown Cooling with RCS level four inches above the Hot Leg centerline. Reactor disassembly has not started. Shutdown Cooling is lost when the only running LPSI pump breaker trips.

How will RCS temperature be monitored to determine a Mode change?

☐ **A** RCS Thot and Tcold Loop RTDs

☐ **B** SDC temperatures RCS to SDC (T351X) and SDC to RCS (T351Y)

☑ **C** Inadequate Core Cooling (ICC) unheated junction thermocouples

☐ **D** Handheld infrared temperature measurements on piping adjacent to RCS to SDC (T351X) and SDC to RCS (T351Y)

Question Misc. Info: New question for the 2018 examination.

Justification

A - INCORRECT: RCS Thot and Tcold Loop RTDs are not correct. These loop RTDs are not used while on SDC with no RCPs operating. The plant would not be operating RCPs while in reduced inventory. The plant Cooldown procedure (2207) specifies when RCS flow is from RCPs or natural circulation only, TAVG is the calculated average of both loop RTDs (two THOTs and two TCOLDs). Also if both SDC and RCPs are operating TAVG is the calculated average of loop RTDs in the operating loop.
PLAUSIBLE: The RCS Thot and Tcold Loop RTDs are used at times to calculate RCS Tavg. Tavg is the temperature of record when determining Mode changes. Per OP 2207, Plant Cooldown procedure, definition 2.4.5; when RCS flow is from RCPs or natural circulation only, TAVG is the calculated average of both loop RTDs (two THOTs and two TCOLDs).

B - INCORRECT: SDC temperatures RCS to SDC (T351X) and SDC to RCS (T351Y) is not correct. The Loss of Shutdown Cooling procedure AOP 2572, caution prior to step 3.7 states; with no SDC flow, T351X and T351Y do not provide an accurate indication of RCS temperature. Accurate indication is provided by UJTEM8-A, UJTEM8-B, UJTEM7-A, UJTEM7-B, and CETs (PPC). The associated step specifies; IF available, MONITOR RCS level and temperature by use of PPC ICC level and temperature display, using unheated junction thermocouples in contact with RCS inventory for temperature indication.
PLAUSIBLE: The SDC temperatures RCS to SDC (T351X) and SDC to RCS (T351Y) are what we normally use to monitor temperature when SDC is in service. The Plant Cooldown procedure, OP 2207 states that TAVG is calculated as the average of "RCS to SDC temperature, T351X" and "SDC to RCS temperature, T351Y" when SDC is in operation with no RCPs operating, SDC flow is greater than 1,000 gpm and T351Y is below Tcold. In mid-loop operation, as the stem states, RCPs would not be operating so we would not use these SDC temperatures to determine TAVG.

C - CORRECT: Inadequate Core Cooling (ICC) unheated junction thermocouples is correct. The Loss of Shutdown Cooling procedure, AOP 2572 specifies under the initial actions for a Loss of SDC, in step 3.7, that IF available, MONITOR RCS level and temperature by use of PPC ICC level and temperature display, using unheated junction thermocouples in contact with RCS inventory for temperature indication.

D - INCORRECT: Handheld infrared temperature measurements on piping adjacent to RCS to SDC (T351X) and SDC to RCS (T351Y) is not correct. This method of determining temperatures in a Loss of SDC if only necessary if ICC temperature monitoring is unavailable (per the stem reactor disassembly has not started therefore ICC is still available) or if electrical bus VR-11 is deenergized (the stem provides no indication that VR-11 is deenergized so it is energized). Therefore the use of Handheld infrared temperature measurements on piping adjacent to RCS to SDC (T351X) and SDC to RCS (T351Y) is not necessary.
PLAUSIBLE: Handheld infrared temperature measurements on piping adjacent to RCS to SDC (T351X) and SDC to RCS (T351Y) is a method that must be used in the Loss of SDC procedure when certain conditions exist. Those conditions are when instrument electrical bus VR-11 is lost and ICC temperature monitoring is unavailable. Both instrument electrical bus VR-11 and ICC temperature monitoring are available so the use of the Handheld infrared temperature measurements on piping adjacent to RCS to SDC (T351X) and SDC to RCS (T351Y) is not necessary but is plausible.

References

Reference (not hand outs)
1. AOP 2572, Loss of Shutdown Cooling, R014-000, page 5, 6, 33, & 34.
2. OP 2207, Plant Cooldown, R042, page 11.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 025 000025 (APE 25) Loss of Residual Heat Removal System / 4

Number AA1.12 **RO** 3.6 **SRO** 3.5 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

AA1.12 Ability to operate and / or monitor the following as they apply to the Loss of Residual Heat Removal System: RCS temperature indicators

Question #: **7** ***Question ID:*** ***2018002*** **Rev.** **0** ☐ **Student Handout?** ☐ **Lower Order?**

Last Edited: 6/20/2018 10:30:11 PM ☑ ***RO*** ☐ ***SRO*** ***Origin:*** ***New*** ☐ **Past NRC Exam?**

Question #: **8** **Question ID: 5000007** Rev. 1 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/1/2018 10:06:58 AM ☑ **RO** ☐ **SRO** **Origin: Bank** ☐ Past NRC Exam?

The following conditions exist on Unit 2:

- The reactor is shutdown
- Both trains of SDC are in service
- RCS temperature is 280°F
- RCS pressure is 160 psia
- RBCCW surge tank level is decreasing

What leak location will produce these indications?

☐ **A** Letdown Heat Exchanger

☐ **B** RCP Thermal Barrier Heat Exchanger

☐ **C** Shutdown Cooling Heat Exchanger

☑ **D** RBCCW Heat Exchanger

Question Misc. Info: MP2 [026 RBC-00-C RO7], NRC-2005

Justification

A - WRONG; Letdown pressure on letdown heat exchanger is greater than RBCCW pressure (160 psia vs 120 psig).
PLAUSIBLE: Student may not understand that letdown pressure on letdown heat exchanger is greater than RBCCW pressure.

B - WRONG; RCS pressure on thermal barrier heat exchangers is greater than RBCCW pressure on the same heat exchangers (160 psia vs 120 psig).
PLAUSIBLE: Student may not understand that RCP seal bleedoff pressure through the thermal barrier is greater than RBCCW pressure.

C - WRONG; With both trains of shutdown cooling in service, the SDC system pressure in the SDC HXs (~ 165 psig) exceeds that of RBCCW (~120 psig).
PLAUSIBLE: Student may not understand that SDC pressure is greater than RBCCW pressure.

D - CORRECT; Service water pressure on RBCCW heat exchanger is less than RBCCW pressure (~45 psig vs ~120 psig). Maximum Service Water pump delta-P of 65 psid in surveillance procedure data sheet (SP-2612A).

References

AOP 2564, R5C0, Table of Potential Leak Paths; Picture of C-06/7 RBCCW Pressure and Flow indication.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 026 000026 (APE 26) Loss of Component Cooling Water / 8

Number AA1.02 **RO** 3.2 **SRO** 3.3 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

AA1.02 Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: Loads on the CCWS in the control room

Question #: **9** **Question ID: 2018010** **Rev.** 1 ☐ **Student Handout?** ☐ **Lower Order?**

Last Edited: 6/15/2018 1:14:03 PM ☑ **RO** ☐ **SRO** **Origin: Bank** ☐ **Past NRC Exam?**

Assume two hypothetical plant scenarios.
In Scenario #1, an RCS leak has emptied the pressurizer, but level is now recovering and has risen to 25%.
In Scenario #2, pressurizer level has been stable at 40%, when a sudden RCS temperature change causes level to rise to 65%.

How and why would the RCS pressure rise differ under these two conditions, based on the given pressurizer level rise?

☐ **A** Scenario #1 has a GREATER rise in pressure, due to the expansion of the water in the pressurizer as it is being heated.

☐ **B** Scenario #1 has a GREATER rise in pressure, due to the RCS temperature rise as Natural Circulation flow develops.

☐ **C** Scenario #1 has a SMALLER rise in pressure, due to the collapse of steam voids in the unaffected steam generator tubes.

☑ **D** Scenario #1 has a SMALLER rise in pressure, due to all the water rising in the pressurizer being at Hot Leg temperature.

Question Misc. Info: LOIT, E32-01-C, LOCA, 2532, PZR, MB-05939

Justification

A - WRONG; As the water is heated, it will flash to steam resulting in raising RCS pressure. However, the PZR water is well below saturation temperature. Therefore it must first be raised in temperature to saturation, then additional heater energy is required due to the latent heat of vaporization.
Plausible; This is credible because when water is heated it will normally expand, which would add to the rate of pressure rise.

B - WRONG; Thot will rise as NC develops, but under normal conditions, it will never reach the saturation temperature of the water that was lost from the pressurizer.
Plausible; The RCS Thot will rise substantially as NC develops and the student may be thinking of the effect of the rise in Thot if the PZR bottomed out at 25% in both cases.

C - WRONG; Although the loss of forced circulation and the PZR empting could cause a steam void in the reactor head, any steaming of the SGs to maintain RCS temperature and establish NC flow would prevent steam voids from forming in the SG tubes.
Plausible; Student may consider that the initial drop in RCS pressure as the PZR emptied could cause steam voids to develop in the SG tubes of any SG not being used for heat removal, which is an expected condition that must be accounted for in other accident scenarios.

D - **CORRECT**; If the PZR has emptied, then all of the water rising up in the PZR comes from Thot, which would be well below saturation for the RCS in the conditions given and would quench the existing steam bubble in the PZR as level increased. Conversely, if the insurge into the PZR starts with level at 25%, then the colder water coming into the PZR is below the volume of water that is at saturation. As the saturated water is pushed up, it will actually superheat the steam bubble through compression, causing a much greater rise in pressure than if the water was 50℉ subcooled (NOP/NOT conditions). With the PZR recovering from empty, for the additional energy added through compression to have a similar effect, the water temperature must first be raised almost to saturation by the PZR heaters. This effectively creates the delay in the rise of PZR pressure.

References

PLC-01-C, R5C0, Pg. 7, 15 and 16

NO Comments or Question Modification History at this time.

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

Number AK3.04 **RO** 2.8 **SRO** 3.3 **CFR Link** (CFR 41.5, 41.10 / 45.6 / 45.13)

AK3.04 Knowledge of the reasons for the following responses as they apply to the Pressurizer Pressure Control Malfunctions: Why, if PZR level is lost and then restored, that pressure recovers much more slowly.

Question #: **10** **Question ID: 2018005** **Rev. 0** ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/14/2018 8:47:47 AM ☑ **RO** ☐ **SRO** **Origin: New** ☐ Past NRC Exam?

The plant is at 100% power, steady state, when a grid disturbance causes the main turbine to trip.

Which of the following conditions are required for the ATWS Mitigation Circuit (i.e.; Diverse SCRAM System) to trigger an Auto Auxiliary Feedwater Actuation (AFAS) in less than 20 seconds?

☐ **A** Low Steam Generator Level.
AND
High Control Channel NI power.

☐ **B** Low Steam Generator Level.
AND
High Safety Channel NI Power.

☑ **C** High Pressurizer Pressure.
AND
High Control Channel NI power.

☐ **D** High Pressurizer Pressure.
AND
High Safety Channel NI power.

Question Misc. Info: MP2*LOIT*3061 [061 AFW-01-C 2530] (8/19/96) ATWS, 2322, AFW, APP, NRC-2008 , NRC-2016

Justification

A - WRONG; Low SG level will trigger an AFAS, but NOT in under 20 seconds. SG level triggers the 205 second TD.
PLAUSIBLE: If the student confuses the "normal" AFAS trigger with the one for the DSS.

B - WRONG; The DSS requires Low SG level OR High pressure and High NI power, but on the Control Channels.
PLAUSIBLE; The student may remember SG level triggers an AFAS, and NI power is required for the DSS to trigger AFAS earlier, but assume the two combined would trigger the actuation earlier.

C - CORRECT; The AFAS has a time delay to trigger on low SG level of 3 minutes and 25 seconds. However, if the DSS senses a high RCS pressure (>2400 psia) combined with NI Control Channel power > 20%, the time delay to trigger is reduced to 10 seconds.

D - WRONG; The DSS gets its NI power input from the Control Channels, NOT the Safety Channels.
PLAUSIBLE; The student may logically assume that with Safety Channels being used for sensing both low SG level and high PZR pressure, that Safety Channels would also be used for sensing high NI power.

References

ARP 2590C-101, R0C0 (C-04, D-13)

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 029 000029 (EPE 29) Anticipated Transient Without Scram / 1

Number EA2.01 **RO** 4.4 **SRO** 4.7 **CFR Link** (CFR 43.5 / 45.13)

EA2.01 Ability to determine or interpret the following as they apply to a ATWS: Reactor nuclear instrumentation

Question #: **11** **Question ID: 8054019** Rev. 1 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/19/2018 1:29:08 PM ☑ **RO** ☐ **SRO** **Origin: Bank** ☐ Past NRC Exam?

A Steam Generator Tube Rupture has occurred. All systems operated normally on the plant trip. EOP 2525, Standard Post Trip Actions, has been completed and EOP 2534, Steam Generator Tube Rupture, is being implemented. The RCS cooldown to Thot less than 515°F has just been completed.

How would the cooldown have been different if the RSST was lost on the plant trip?

☑ **A** SG pressure should be lower in both of the SGs because of the larger RCS Delta-T required when cooling down without the RSST available.

☐ **B** SG pressure should be lower in the ruptured SG because the SGs are no longer linked by the Main Steam header and the ruptured SG will be depressurized by RCS leakage.

☐ **C** SG pressure should be the same on the cooldown, with or without the RSST available, because the cooldown is always to the same RCS temperature of 515°F.

☐ **D** SG pressure should be higher in the ruptured SG because of the larger RCS Delta-T required when cooling down without the RSST available.

Question Misc. Info: MP2*LORT CEN-152, SGTR, 2534, NRC-2008

Justification

A - CORRECT: SG pressure should be lower in both of the SGs because of the larger RCS Delta-T required when cooling down without the RSST available is correct. The loss of the RSST would remove power from the RCPs. With no RCPs operating RCS heat removal would be from Natural Circulation. Loop delta temperatures (Tc and Th) on a normal trip with RCPs operating is approximately 2 °F. Loop delta temperatures (Tc and Th) on a normal trip without RCPs operating are approximately 20-25 °F. The cooldown for a SGTR lowers RCS Th to less than 515 °F. A cooldown without RCPs would require Tc to be approximately 20 °F lower than a cooldown with RCPs. Tc, with RCPs operating, after cooling down to a Th of less than 515 °F would be around 513 °F. Tc, without RCPs operating, after cooling down to a Th of less than 515 °F would be around 495 °F, due to the greater delta temperature required for heat removal at the lower mass flow rate. Tc is directly tied to steam generator pressure. Lower Tc yield lower steam generator pressures since the steam generator secondary side is in a saturated state. From the Steam Tables, a Tc of 512 °F corresponds to a steam generator pressure of 757 psia and a Tc of 496 °F corresponds to a steam generator pressure of 656 psia.

B - INCORRECT: SG pressure should be lower in the ruptured SG because the SGs are no longer linked by the Main Steam header and the ruptured SG will be depressurized by RCS leakage is not correct. During the cooldown both steam generators are cooled the same amount by lowering steam pressure with the condenser dump valves or ADVs. Therefore steam generator pressures are maintained at roughly the same pressure until the ruptured steam generator is isolated. If cooled down with the condenser dump valves the steam remain linked through the main steam header.
PLAUSIBLE: The examine may think that RCS leakage into the SG through a ruptured tube could act like "spray flow" into the pressurizer, causing SG pressure to drop more than expected. However, this effect is NOT possible when steam generator level is maintained in above the tubes as required by procedure.

C - INCORRECT: SG pressure should be the same on the cooldown, with or without the RSST available, because the cooldown is always to the same RCS temperature of 515 °F is not correct. With the RSST there is forced circulation with the RCPs and RCS delta temperatures are approximately 2 °F. Without the RSST the RCP do not have power and RCS heat removal is by Natural Circulation. The RCS delta temperatures in Natural Circulation are in the order of 20-25 °F. To obtain a cooldown to a Thot of less than 515 °F, in Natural Circulation, the Tcold must be approximately 20 °F lower than a forced circulation cooldown. Tc is directly tied to steam generator pressure. Lower Tc yield lower steam generator pressures since the steam generator secondary side is in a saturated state. Therefore steam generator pressures can't be the same when Thot reaches 515 °F for forced and Natural Circulation cooldowns.
PLAUSIBLE: The examinee may think that steam generator pressure is linked to RCS Thots and not Tcolds. As such the steam generator pressure will not change since the Thots are the same. They could miss that the loss of the RSST removes the RCPs from operation. They could miss the significant change in RCS loop deltas from forced flow to Natural Circulation flow.

D - INCORRECT: SG pressure should be higher in the ruptured SG because of the larger RCS Delta-T required when cooling down without the RSST available is not correct. The two loops will remain coupled as both steam generators are being used for the cooldown. Therefore, both loop Tcold temperatures should be about the same. Any differences in SG pressures would be based solely on the throttled position of the individual ADVs during the cooldown.
PLAUSIBLE: The examinee may think that steam generator pressure would be higher in the ruptured steam generator due to the higher pressure RCS leaking into it.

References

E34-01-C Steam Generator Tube Rupture Lesson Text, R4

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 038 000038 (EPE 38) Steam Generator Tube Rupture / 3

Question #: **11** **Question ID: 8054019** **Rev. 1** ☐ **Student Handout?** ☐ **Lower Order?**

Last Edited: 6/19/2018 1:29:08 PM ☑ **RO** ☐ **SRO** **Origin: Bank** ☐ **Past NRC Exam?**

Number EK1.03 **RO** 3.9 **SRO** 4.2 **CFR Link** (CFR 41.8 / 41.10 / 45.3)

EK1.03 Knowledge of the operational implications of the following concepts as they apply to the SGTR: Natural circulation

Question #: **12** **Question ID: 2018003** Rev. 0 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/24/2018 10:53:14 AM ☑ RO ☐ SRO **Origin: New** ☐ Past NRC Exam?

The plant has experienced an Excess Steam Demand event inside Containment. The Operator will take action by controlled steaming of the least affected steam generator using the Atmospheric Dump valve when ____1____ to ____2____ to prevent Pressurizer Thermal Shock (PTS).

☐ **A** 1. Narrow Range Steam Generator level reaches 0 inches
2. Stabilize RCS temperature to limit expansion of the RCS

☑ **B** 1. CET temperatures start to rise
2. Stabilize RCS temperature to limit expansion of the RCS

☐ **C** 1. Narrow Range Steam Generator level reaches 0 inches
2. Control restoration of RCS temperature to normal post- trip value

☐ **D** 1. CET temperatures start to rise
2. Control restoration of RCS temperature to normal post- trip value

Question Misc. Info: New question for the 2018 examination.

Justification

B - CORRECT: In an Excess Steam Demand event inside Containment, the operator will take action by controlled streaming of the least affected steam generator using the Atmospheric Dump valve when CET temperatures start to rise to stabilize RCS temperature to limit expansion of the RCS to prevent Pressurized Thermal Shock (PTS).

EOP 2536, step 11 states that WHEN the most affected steam generator has boiled dry, as indicated by CET temperatures rising, STABILIZE CET temperature by controlled steaming of the least affected steam generator using the ADV. EOP 2525 under step 7 provides the same guidance. This is done to prevent the RCS from expanding due to an uncontrolled temperature rise. The RCS inventory will have been increased by Charging and possibly HPSI. If the RCS is allowed to return to normal post reactor trip temperature, pressure would rise rapidly and result in PTS.

A - INCORRECT: Narrow Range Steam Generator level reaches 0 inches and Stabilize RCS temperature to limit expansion of the RCS is not correct because action is taken based on CET temperatures not steam generator level. Steam generator level is not used because at 0 inches indicated level there is still heat removal. This is because level indication does not go to the bottom of the steam generator secondary side.
PLAUSIBLE: Using Narrow Range Steam Generator level is plausible because when level is lost the RCS temperature will start to rise because there will be no heat removal. The examinee may reason that if there is no level then there is no heat removal; which would be correct except at 0 inches indicated there is still some inventory in the steam generator secondary side. Also prior procedure revisions did use steam generator level as the indication for when action is taken with the ADV. This was changed from steam generator level to CET when it was determined that CETs would be a better indication due to water remaining in the steam generators at 0 inches.

C - INCORRECT: Narrow Range Steam Generator level reaches 0 inches and Control restoration of RCS temperature to normal post- trip value is not correct because action is taken based on CET temperatures not steam generator level. Steam generator level is not used because at 0 inches indicated level there is still heat removal. This is because level indication does not go to the bottom of the steam generator secondary side. Control restoration of RCS temperature to normal post- trip value is not correct because the procedure has you stabilize temperature to prevent the additional inventory added to not expand.
PLAUSIBLE: Using Narrow Range Steam Generator level is plausible because when level is lost the RCS temperature will start to rise because there will be no heat removal. The examinee may reason that if there is no level then there is no heat removal; which would be correct except at 0 inches indicated there is still some inventory in the steam generator secondary side. Control restoration of RCS temperature to pre-event value is plausible because for an ESD downstream of the MSIVs, where the cooldown is terminated by MSIV closure the guidance in EOP 2525 is to stabilize temperature and then return temperature to the normal post trip temperature (this guidance is contained in EOP 2525, step 7.c.4)

D - INCORRECT: CET temperatures start to rise and Control restoration of RCS temperature to normal post- trip value is not correct. CET temperatures start to rise is correct but control restoration of RCS temperature to normal post- trip value is not correct. Control restoration of RCS temperature to normal post- trip value is not correct because the procedure has you stabilize temperature to prevent the additional inventory added to not expand.
PLAUSIBLE: Control restoration of RCS temperature to pre-event value is plausible because for an ESD downstream of the MSIVs, where the cooldown is terminated by MSIV closure the guidance in EOP 2525 is to stabilize temperature and then return temperature to the normal post trip temperature (this guidance is contained in EOP 2525, step 7.c.4)

References

EOP 2525 Standard Post Trip Actions

Question #: **12** **Question ID: 2018003** **Rev. 0** ☐ **Student Handout?** ☑ **Lower Order?**

Last Edited: 6/24/2018 10:53:14 AM ☑ **RO** ☐ **SRO** **Origin: New** ☐ **Past NRC Exam?**

CEN 152 Emergency Procedure Guidelines

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** E05 000040 (APE 40; CE E05) Steam Line Rupture Excessive Heat Transfer / 4

Number EA2.2 **RO** 3.4 **SRO** 4.2 **CFR Link** (CFR: 43.5 / 45.13)

EA2.2 Ability to determine and interpret the following as they apply to the (Excess Steam Demand): Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.

Question #: **13** **Question ID: 54198** **Rev. 1** ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/14/2018 9:05:27 AM ☑ **RO** ☐ **SRO** **Origin: Bank** ☐ Past NRC Exam?

Unit 2 has tripped due to a loss of offsite power.
The following conditions exist:

- "A" Aux. Feedwater pump OOS due to planned maintenance.
- "B" Aux. Feedwater pump ran for 5 minutes and shutdown due to a seized bearing.
- Turbine driven Aux. Feedwater (TDAFW) pump tripped due to failed over speed mechanism.
- #1 SG level is 10% and slowly lowering.
- #2 SG level is 20% and slowly lowering.
- The crew has begun implementing EOP 2537, Loss Of All Feedwater
- Maintenance is working to restore a source of feedwater and expects to be able to repair the TDAFW pump.
- All other plant equipment is functioning as designed.

Which one of the following conditions would require Once-Through-Cooling to be immediately initiated, IAW EOP 2537?

☑ **A** Cold leg temperatures have risen 10°F uncontrollably.

☐ **B** #1 SG wide range is 70", #2 SG wide range level is 195".

☐ **C** CET Temperatures are 10°F above Hot Leg temperatures.

☐ **D** Either SG level is < 33% with less than normal feed flow.

Question Misc. Info: MP2*LORT*5696 2537, LOAF, OTC

Justification

"A" - CORRECT; OP-2537, Contingency Actions require use of Once-Through-Cooling as a success path for either of the following:

- One SG level is </= 70" and the other SG level is </= 165"
  OR
- **RCS Tcold rises uncontrollably by >/= 5 °F.**

"B" - WRONG; #2 SG level would also have to be </= 165" under these conditions.
PLAUSIBLE; If any HPSI, PORVs or ADVs are unavailable, then OTC should be initiated <u>prior</u> to one SG </= 70" and the other SG </= 165". The student may confuse the equipment problem requirements with actual initiation requirements.

"C" - WRONG; This condition is indicative of unsatisfactory Natural Circulation flow, but is not a direct requirement for OTC initiation.
PLAUSIBLE; The student may recognize that a LOOP will require NC flow be verified. The >10°F difference between CETs and Th is an indication that the SGs are not being steamed enough. The student may consider a problem with NC flow caused by poor SG heat removal as grounds for initiating OTC.

"D" - WRONG; The step in EOP 2537 immediately following the OTC requirements discusses actions for abnormally low SG level if < 33%. It requires a very controlled increase in feedwater flow, but is not a requirement for OTC.
PLAUSIBLE; The student may recall that AOP 2537 has an action step based on SG level abnormally low at < 33% and remember that level triggered action as one involving OTC.

References

EOP 2537, R25C0, St. 6 Contingency Actions

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** E06 000054 (APE 54; CE E06) Loss of Main Feedwater /4

Number EK2.2 **RO** 3.5 **SRO** 4.0 **CFR Link** (CFR: 41.7 / 45.7)

Knowledge of the interrelations between the (Loss of Feedwater) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Question #: **14** | ***Question ID*: 2018006** | **Rev.** 0 | ☐ **Student Handout?** | ☑ **Lower Order?**

Last Edited: 6/14/2018 9:05:46 AM | ☑ ***RO*** | ☐ ***SRO*** | ***Origin*: New** | ☐ **Past NRC Exam?**

EOP 2530, Station Blackout, requires power to a vital AC bus be restored, so that which of the following can be completed within 60 minutes?

☐ **A** Start an Instrument Air Compressor to restore pressurizer sprays and monitor RCS pressure.

☐ **B** Start a Boric Acid Pump to restore boric acid injection and monitor wide range nuclear instruments.

☐ **C** Start a Service Water pump to restore DC Room Ventilation and monitor DC room temperatures.

☑ **D** Start the selected Charging Pump to restore RCS inventory and monitor pressurizer level.

Question Misc. Info: MP2*ILT 2530, Blackout, Battery

Justification

A - WRONG; The accident analysis for a Station Blackout assumes Unit 3 is still available to provide either Instrument Air (IA) or AC power. Also, backup air bottles exist to provide IA to the Aux. Spray Valve for control of PZR spray if IA is totally lost.
PLAUSIBLE; Student may remember that a total loss of IA would prevent the opening of any PZR spray valve for RCS pressure control and IA is the very first thing the EOP directs be restored after recovering AC power.

B - WRONG; Boric acid injection is not immediately required because during a Station Blackout event, the required SDM drops to 1% and no other accident is assumed.
PLAUSIBLE; Student may assume that without a source of BA injection and the probable need for an RCS cooldown, SDM will be challenged.

C - WRONG; Although important for proper VIAC Inverter function, DC room ventilation is not required to be implemented immediately following the restoration of AC power.
PLAUSIBLE; High switchgear room temperature will have a negative impact on the operation of the VIAC inverters, which could affect control room instrumentation. Student may not remember the exact sequence of steps in the EOP and the restoration of switchgear room cooling and service water flow (it's heat sink) immediately follows the direction to start a charging pump.

D - CORRECT; Unit 2 is considered an 8 hour coping plant because the core is not expected to uncover for 8 hours following a loss of all AC. However, this is on the assumption that a charging pump is started within one hour of the event initiation.

References

OP 2260, R15C0, EOP 2530 TCOA #1

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 055 Loss of Offsite and Onsite Power (Station Blackout)

Number EA1.07 **RO** 4.3 **SRO** 4.5 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to operate and monitor the following as they apply to a Station Blackout: Restoration of power from offsite

Question #: **15** **Question ID:** 8054131 **Rev.** 1 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/14/2018 9:05:54 AM ☑ **RO** ☐ **SRO** **Origin:** Bank ☐ Past NRC Exam?

The plant is on Shutdown Cooling using the "B" LPSI pump. Then, the RSST supply breaker to bus 24D trips open on a breaker failure.

Which one of the following describes the automatic response of the "B" LPSI Pump?

☐ **A** Because of the anti-pumping circuit, the ESAS undervoltage signal must be reset to restart the "B" LPSI pump.

☐ **B** Because the "fingers" are open on C07R the "B" LPSI pump will immediately restart the moment 24D is reenergized.

☐ **C** Because SDC is in use, the "B" LPSI pump will automatically restart on the applicable sequence step.

☑ **D** Because there is no SIAS present, the "B" LPSI pump must be manually restarted once 24D is reenergized.

Question Misc. Info: MP2*LORT 2310, 2572, SDC, LNP, NRC-2008

Justification

A - WRONG; Although the LPSI pump has an "anti-pump" circuit, it is only armed if an automatic start signal is present and the pump is then shutdown by a signal other than a load shed. Therefore, it is NOT interlocked off under these conditions.
PLAUSIBLE; The student may misinterpret the interrelationship between the Load Shed signal and the Anti-Pump interlock. This often occurs with the Aux. Feedwater Pumps, if they are turned off with an AFAS signal present in preparation for re-energizing their dead bus. When the vital bus is reenergized in this situation, the AFW pump CANNOT be restarted until the AFAS signal is reset.

B - WRONG; The "fingers" on C07R do NOT affect the load shed signal, therefore the pump will still be turned off and must be manually restarted. The C07R fingers prevent tripping of the LPSI pumps due to Main Turbine testing or 345 kV breaker operations.
PLAUSIBLE; The student may recall that the fingers on C07R are designed to prevent inadvertent tripping of the LPSI pump due to various breaker control circuit operations and assume this will allow the pump to restart normally.

C - WRONG; Even though the LPSI pump handswitch is in the "Normal-After-Closed" position, there is no constant "breaker close" or start signal present for the pump. Only a ESAS SIAS would send a start signal to the LPSI pump.
PLAUSIBLE; The student may recall that in all other situations (accident) where the LPSI pump is running, it is automatically restarted by the sequencer. In the SDC function, the LPSI pump is being used as the only means of core heat removal, leading one to believe it is acting in a similar role as in accident mitigation.

D - CORRECT; There is no automatic restart of the LPSI pumps on an LNP, but they do get a load shed signal. Because of the load shed signal, the pump will not restart when power is restored, even though the control board handswitch is still in the Normal-After-Close position. Therefore, to restore SDC flow, the LPSI pump must be manually restarted after power is restored to the vital bus.

References

LP SDC-00-C, R6C1, Pg. 39-40

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 056 000056 (APE 56) Loss of Offsite Power / 6

Generic K/A Selected

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

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

G2.4.9 Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Question #: **16** **Question ID: 2018008** Rev. 0 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/24/2018 10:56:35 AM ☑ RO ☐ SRO **Origin: New** ☐ Past NRC Exam?

A plant trip has occurred as a result of the loss of Vital 125 VDC Instrument Panel DV20. Of the equipment listed below, what is the full complement of equipment available for the operator to control and monitor, the supply of Auxiliary Feedwater to the Steam Generators, from Main Control Board C05.

- Turbine Driven Auxiliary Feedwater (TDAFW) pump
- "A" Motor Driven Auxiliary Feedwater (MDAFW) pump
- "B" Motor Driven Auxiliary Feedwater (MDAFW) pump
- Auxiliary Feedwater (AFW) flow indication to the #1 Steam Generator (SG)
- Auxiliary Feedwater (AFW) flow indication to the #2 Steam Generator (SG)
- "A" Auxiliary Feedwater Regulating valve (AFRV)
- "B" Auxiliary Feedwater Regulating valve (AFRV)

☐ **A** "A" MDAFW pump, "A" AFRV, and AFW flow indication to both SGs

☑ **B** TDAFW pump, "A" MDAFW pump, "A" AFRV, and AFW flow indication to both SGs

☐ **C** TDAFW pump, "A" MDAFW pump, "A" AFRV, and AFW flow indication to #1 SG

☐ **D** "A" MDAFW pump, "B" MDAFW pump, "A" AFRV, and AFW flow indication to #1 SG

Question Misc. Info: New question for the 2018 examination.

Justification

A - INCORRECT: "A" MDAFW pump, "A" AFRV, and AFW flow to both SGs is not correct. It is not correct because it is not the full complement of equipment and indication available at C05. All in the answer are correct but the TDAFW pump is also available.
PLAUSIBLE: It is plausible because the TDAFW pump is powered from Facility 2 normally. The examinee might not remember that there are key switches to transfer TDAFW pump control power from Facility 2 to Facility 1. They could also assume that only Facility 1 equipment is available since DV20 is facility 2. They could also not remember that AFW flow is powered from VA10 and VA20 and not DV power.

B - CORRECT: TDAFW pump, "A" MDAFW pump, "A" AFRV, and AFW flow to both SGs is correct. The TDAFW pump can be controlled at C05 by swapping the power supply for the TDAFP steam valve and speed control power to DV10 (step 2.c of 2506B). The "A" MDAFW is not affected by a loss of DV20. The "A" MDAFW pump is powered from vital 4160 VAC with breaker control from DV10. The "A" AFRV is not affected by a loss of DV20. The "A" AFRV is powered by DV10 and VA10. The AFW flow indication to both SGs is not affected by a loss of DV20. The AFW flow indication to both SGs is powered from VA10 and VA20. Both SGs have two flow indicators; one powered from VA10 and the other powered from VA20. Control of the "B" MDAFW pump is not available at C05 because it has no breaker control power. DV20 is the breaker control power for the "B" MDAFW pump. Control of the "B" AFRV at C05 is not available and the valve fails full open.

C - INCORRECT: TDAFW pump, "A" MDAFW pump, "A" AFRV, and AFW flow to #1 SG is not correct. It is not correct because it is not the full complement of equipment and indication available at C05. All in the answer are correct but flow to the #2 SG also available. AFW flow is powered from VA10 and VA20 and not DV power.
PLAUSIBLE: The examinee could not remember that AFW flow is powered from VA10 and VA20 and not DV power.

D - INCORRECT: "A" MDAFW pump, "B" MDAFW pump, "A" AFRV, and AFW flow to #1 SG is not correct. The answer does not include the TDAFW pump and includes the "B" MDAFW pump which is not available. Control of the "B" MDAFW pump is not available at C05 because it has no breaker control power. DV20 is the breaker control power for the "B" MDAFW pump. Flow to the #2 SG also available. AFW flow is powered from VA10 and VA20 and not DV power.
PLAUSIBLE: The examinee may understand that DV20 affects the facility 2 side which is the #2 SG. Both the "A" and "B" MDAFW pumps are on the facility 1 side of the Auxiliary Feedwater system (Facility 1 and 2 are separated by FW-44). The examinee could reason that since the "B" MDAFW pump is on the Facility 1 side of the AFW system that it is powered from Facility 1. The examinee could also not remember that AFW flow is powered from VA10 and VA20 and not DV power.

Question #: **16** **Question ID: 2018008** **Rev.** 0 ☐ **Student Handout?** ☐ **Lower Order?**

Last Edited: 6/24/2018 10:56:35 AM ☑ **RO** ☐ **SRO** **Origin: New** ☐ **Past NRC Exam?**

References

AOP 2506B Loss of Vital 125 VDC Instr Panel DV20, R 003-00
AOP 2504C Loss of 120 VAC Vital Instr Panel VA-10, R 005-00
AOP 2504D Loss of 120 VAC Vital Instr Panel VA-20, R 005-00

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 058 000058 (APE 58) Loss of DC Power / 6

Number AA2.03 **RO** 3.5 **SRO** 3.9 **CFR Link** (CFR: 43.5 / 45.13)

AA2.03 Ability to determine and interpret the following as they apply to the Loss of DC Power: DC loads lost; impact on ability to operate and monitor plant systems

Question #: **17** ***Question ID:*** ***78720*** **Rev.** 4 ☐ **Student Handout?** ☐ **Lower Order?**

Last Edited: 6/14/2018 9:07:21 AM ☑ ***RO*** ☐ ***SRO*** ***Origin:*** ***Bank*** ☐ **Past NRC Exam?**

The plant is at 100% power, All conditions are normal, when 480 VAC load center bus 22E is lost.

Per AOP 2503E, the following alignments have been made:

- 24E is powered from Unit 3 bus 34B.
- "B" Service water pump is in service supplying the "A" Service Water Header.
- "B" Service Water Strainer is being powered from bus 22F.

Based upon the given conditions, which one of the following describes the impact on the Service Water System Tech. Spec. 3.7.4.1 LCO?

☐ **A** Not applicable because both Service Water loops are still functioning as designed.

☑ **B** Is applicable because Facility 1 Service Water is not OPERABLE while its strainer is being powered by Facility 2.

☐ **C** Not applicable because Facility 1 Service Water strainer is still functioning as designed.

☐ **D** Is applicable because Facility 2 Service Water is not operable while a Facility 1 electrical load is cross tied to it.

Question Misc. Info: MP2*ILT AOP 2503, TS, TSAS, SW, Service Water

Justification

A - WRONG; Tech. Specs. requires the pump and the pump's strainer be powered from the same facility.
PLAUSIBLE; Student may believe the AOP would align the equipment to ensure both Service Water (SW) headers are functioning to maintain the plant **at-power**, when in reality the basis for the action is to prevent a plant trip and allow for a controlled plant shutdown.

B - CORRECT; The "A" SW pump strainer is powered from 22E, which was lost. The "B" SW pump is powered from 24E, which is aligned to get its DC control power from whichever facility it is electrically tied to (in this case, 24C or Fac. 1). However, the "B" SW pump strainer can only get its power from either 22E or 22F. In this scenario, the loss of 22E requires the "B" strainer be powered from Facility 2 ("B" Train) power (22F) to allow it to continue to function.

C - WRONG; The SW pump strainer's ability to function does NOT in an of itself ensure Tech. Spec. Operability. It must still be powered from the same facility as the associated pump.
PLAUSIBLE; Student may remember the basic reason for the abnormal power alignment directed by the AOP was to ensure the SW strainer aligned for Facility 1 would still function, but forget the underlying issue is with the limitations of the "B" (swing) SW pump strainer control circuitry.

D - WRONG; Facility 2 SW fully meets Tech. Spec. Operability even with a Facility 1 load ("B" strainer) tied to it. This is because the strainer could normally be tied to it if the "B" SW pump was being used in place of the "C" SW pump and both 480 VAC load centers, 22E and 22F, were energized. In such an instance, power to the "C" strainer would not be isolated.
PLAUSIBLE; The student may recognize that Facility 2 is indeed carrying the load of two SW strainers, one of which is being used for the other facility. If the SW pump was aligned in a similar fashion, the header would not be operable.

References

AOP 2503E, R04C00, Pg. 8-10

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 062 000062 (APE 62) Loss of Nuclear Service Water / 4

Generic K/A Selected

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

Question #: **17** | ***Question ID:*** ***78720*** | **Rev.** **4** | ☐ **Student Handout?** | ☐ **Lower Order?**

Last Edited: 6/14/2018 9:07:21 AM | ☑ ***RO*** | ☐ ***SRO*** | ***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)

G2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Question #: **18** **Question ID:** **76395** **Rev.** 4 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/1/2018 12:09:17 PM ☑ **RO** ☐ **SRO** **Origin:** **Bank** ☐ Past NRC Exam?

The plant is at 100% power and a degraded voltage condition exists. The crew has entered AOP 2580, Degraded Voltage, and is performing all required actions.

Which one of the following conditions would require the crew to trip the reactor and perform EOP 2525, "Standard Post Trip Actions"?

☑ **A** The voltage remains less than 3,700 volts on the 4.16 kV buses after all attempts are made to correct the degraded voltage condition.

☐ **B** Both Vital 4.16 kV busses 24C and 24D voltage have been less than 3,900 Volts for over an hour.

☐ **C** The crew is unsuccessful in raising bus voltage by reducing all unnecessary plant electrical loads.

☐ **D** The crew raised the Turbine Generator MVARs to the Reactive Capability Curve limits and 4.16 kV bus voltage is still less than 3,900 Volts.

Question Misc. Info: MP2*LORT*2580, AOP, Degraded Voltage

Justification

A - CORRECT; AOP 2580 Step 3.11 states that if voltage remains less than 3,700 volts on the 4.16 kV buses then the reactor is tripped and EOP 2525 is entered.

B - WRONG; At </= 3900 VAC the RSST is considered inoperable by Tech. Specs., but a plant trip is not required.
PLAUSIBLE; Student may confuse the inoperability of the RSST value, which would eventually require a plant shutdown, with the plant trip requirement. When voltage drops to 3900 VAC the AOP states the RSST is no longer considered OPERABLE and directs performance of the offsite power availability surveillance within one hour.

C - WRONG; Although this means the plant is at the mercy of the grid operators, the AOP does not require a plant trip under this condition.
PLAUSIBLE; Student may recognize that the plant is now totally vulnerable to further grid degradation, and , therefore, a trip would be the conservative choice of actions.

D - WRONG; The AOP does not require a plant trip until voltage is </= 3700 VAC.
PLAUSIBLE; This is a combination of "best efforts" and "RSST inoperability", where the student may believe that with the main generator at its limit the only way left to protect plant equipment by divorcing it from the grid.

References

AOP 2580, R03C06, Pg. 10-11

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 077 000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6

Number AK3.02 **RO** 3.6 **SRO** 3.7 **CFR Link** (CFR: 41.4, 41.5, 41.7, 41.10 / 45.8)

AK3.02 Knowledge of the reasons for the following responses as they apply to Generator Voltage and Electric Grid Disturbances: Actions contained in abnormal operating procedure for voltage and grid disturbances

Question #: **19** **Question ID: 2018042** **Rev. 0** ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/14/2018 9:07:23 AM ☑ **RO** ☐ **SRO** **Origin: New** ☐ Past NRC Exam?

The reactor was started up following a refueling outage and brought to 1% power, before it had to be shutdown due to problems with the main turbine.

A reactor startup is now being performed using the Pull-To-Criticality method.
Group 7 CEAs are at 30 steps out and presently being withdrawn, when a steady positive startup rate is achieved. The RO releases the "Insert-Withdraw" switch, but the "withdraw" contacts in the switch stick closed and Group 7 continues to withdraw.

Which of the following would act to protect the core from a sudden power increase?

- ☐ **A** FTC and MTC would work to slow the power rise until a CEA Motion Inhibit stops CEA withdrawal.
- ☐ **B** FTC and MTC would work to slow the power rise until a CEA Withdrawal Prohibit stops CEA withdrawal.
- ☐ **C** Only FTC would work to slow the power rise until a CEA Motion Inhibit stops CEA withdrawal.
- ☑ **D** Only FTC would work to slow the power rise until a CEA Withdrawal Prohibit stops CEA withdrawal.

Question Misc. Info: MP2*ILT FTC, MTC, CWP, RPS

Justification

A - WRONG; Because the CEA Withdrawal Prohibit CWP interlock is triggered by RPS High Power "Pre-Trips", it should to trigger and stop CEA withdrawal before reactor power drives the PDIL into the CEA position.
PLAUSIBLE; Student may remember the CWP as being bypassed below 15% power, like the Local Power Density trip, or remember it as only being triggered by TM/LP pre-trips.

B - WRONG; RCS boron concentration at this time in core life would drive MTC positive.
PLAUSIBLE; Student may have the misconception that bringing the reactor up in power for a day would generate enough xenon to offset the positive MTC boron concentration.

C - WRONG; The CWP interlock is triggered by RPS High Power "Pre-Trips", which would be at their floor value of 14.6% power, well below the level needed for the PDIL to be triggered. The effect of a positive MTC would be dampened by the condenser steam dumps, which would function to limit the rise in RCS temperature.
PLAUSIBLE; Student may feel the weak FTC could not slow power down enough to overcome the positive reactivity added by a rod withdrawal, combined with the positive MTC, to prevent power from overshooting the high power pre-trips and drive the PDIL setpoint into the CEA position.

D - CORRECT; Only FTC is available to slow the power rise because the RCS boron concentration at this time in core life would drive MTC positive. Also, CWP interlock is triggered by RPS High Power "Pre-Trips", which is designed to trigger before an actual trip on high power.

References

ARP 2590C-110, R0C0, CWP and OP 2203, Plant Startup, R28C0, Pg. 22 NOTE

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 001 Continuous Rod Withdrawal

Number AK1.18 **RO** 3.4 **SRO** 3.8 **CFR Link** (CFR 41.8 / 41.10 / 45.3)

AK1.18 Knowledge of the operational implications of the following concepts as they apply to Continuous Rod Withdrawal: Fuel temperature coefficient

Question #: **20** **Question ID: 2018009** Rev. 1 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/11/2018 10:41:18 AM ☑ **RO** ☐ **SRO** **Origin:** **Bank** ☐ Past NRC Exam?

The plant is operating at 100% power when a CEA drops to the bottom of the core.

Which of the following conditions would require the greater amount of turbine load reduction to stabilize RCS temperature?

- ☑ **A** Beginning Of Life due to the overriding effect of the higher RCS boron concentration.
- ☐ **B** End Of Life due to the higher CEA worth from the diminished competition from boron.
- ☐ **C** Beginning Of Life due to burnable poisons adding to the resonance capture of neutrons.
- ☐ **D** End Of Life due to the increased amount of plutonium in the active regions of the core.

Question Misc. Info: MP2*LOIT, AOP, 2556, IH-2011

Justification

A - CORRECT; The higher boron concentration at BOL will result in a very low negative value for MTC. This would translate to requiring a greater RCS temperature drop to add the necessary amount of positive reactivity to compensate for the negative reactivity added by the dropped CEA. Therefore, a larger secondary power reduction is required to stabilize RCS temperature at the new lower reactor power level and restore RCS temperature to program.

"B" - WRONG; The effect from the higher negative value of MTC at EOL easily overrides the higher CEA worth at EOL.
PLAUSIBLE; The student may focus on the true fact that CEAs are worth more at EOL.

"C" - WRONG; Although burnable poisons are only present at BOL, they are designed to capture thermal neutrons in competition with the fresh uranium fuel and not necessarily in the resonance region. Therefore, their neutron absorption effect does not change appreciably with temperature.
PLAUSIBLE; Student may recognize that the burnable poisons due add to neutron capture at BOL and assume there presence has a large enough impact on FTC to required a greater load reduction.

"D" - WRONG; Although the power coefficient increasing over core life is the key, the PC increase is mainly due to the increase in the MTC value, which far exceeds the increase in FTC.
PLAUSIBLE; The student may focus on the fact that higher amounts of plutonium building in over core life do increase the value of FTC (power coefficient), and equate this to the need for a larger drop in RCS temperature to compensate.

References

GFES and AOP 2585, IOA, R3C0, St. 8

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 003 000003 (APE 3) Dropped Control Rod / 1

Number AK1.16 **RO** 2.9 **SRO** 3.2 **CFR Link** (CFR 41.8 / 41.10 / 45.3)

AK1.16 Knowledge of the operational implications of the following concepts as they apply to Dropped Control Rod: MTC

Question #: **21** **Question ID: 8600020** **Rev. 1** ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/14/2018 9:07:26 AM ☑ **RO** ☐ **SRO** **Origin: Bank** ☐ Past NRC Exam?

A plant startup is in progress with reactor power approximately 2% and all equipment functioning normally.

Then, the high voltage power supply to the Channel "A" Wide Range Nuclear Instruments fails, such that the channel is now reading 1 E-6 % power.

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

☐ **A** The Zero Power Mode Bypass will arm on Channel "A".

☑ **B** The Level 1 and Level 2 Bistables will reset on Channel "A".

☐ **C** The PDIL alarm and interlock on CEAPDS is now bypassed.

☐ **D** The Power Trip Test Interlock (PTTI) will arm on Channel "A".

Question Misc. Info: MP2*LOIT NIS, 2202, Failure, NRC-2008

Justification

A - WRONG: A bypass key for the channel must be in the "bypass" position when the Wide Range NI for that channel drops below 1X10-4% in order to arm the Zero Power Mode Bypass. The key is procedurally required to be removed prior to criticality.
PLAUSIBLE; Student may remember that the WRNI dropping below 1X10-4% power will ALLOW initial arming, or re-arming, of the Zero Power Mode Bypass. If the key were left in the bypass position and power dropped below the trigger value, the ZPMB would re-arm automatically.

B - CORRECT: The Level 1/2 bistables will RESET on this channel when the signal drops below 1X10-4% and would set up the circuits that would allow for the conditions described in choices "A" and "C" to occur.

C - WRONG: The PDIL bypass would NOT activate on a single WRNI failing low, as more than one channel below 1X10-4% is required to activate the bypass. However, if the bypass is armed, only one channel failing above 1X10-4% would clear the bypass and re-arm the PDIL interlock.
PLAUSIBLE; The student may confuse the failure mechanism that will "re-arm" the interlock with the one that bypasses it.

D - WRONG: The PTTI interlock would indeed be armed for this channel, IF the failed detector power supply were on a Linear Channel.
PLAUSIBLE; Student may confuse the NI failure mechanism that would trigger a PTTI.

References

LP NIS-01-C, R5C2, Pg. 20

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 032 000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7

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

AK2.01 Knowledge of the interrelations between the Loss of Source Range Nuclear Instrumentation and the following: Power supplies, including proper switch positions

Question #: **22** **Question ID: 2018007** Rev. 0 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/17/2018 11:18:22 AM ☑ RO ☐ SRO **Origin: New** ☐ Past NRC Exam?

Main Condenser vacuum is 2.5 in Hg and getting worse. The Loss of Condenser Vacuum procedure AOP 2574 has been entered.

Which of the following procedure actions will be used to address the degrading condenser vacuum due to a reduction in condenser heat transfer?

- ☐ **A** Ensure steam seal pressure is being maintained between 2 to 6 psig.
- ☐ **B** Throttle open the turbine exhaust hood spray bypass to lower exhaust hood temperatures.
- ☐ **C** Verify the operating SJAE steam supply pressure is 200 - 220 psig.
- ☑ **D** Refer to AOP 2517, Circulating Water Malfunctions, and raise circulating water pump speed.

Question Misc. Info: New question for the 2018 examination.

Justification

This question assesses the knowledge of the examinee for a loss of main condenser vacuum. The examinee must understand what equipment is available to address a loss of vacuum. They must understand what could cause a loss of vacuum and the equipment available to address each of the different causes.

A - INCORRECT: Ensure steam seal pressure between 2 to 6 psig is not correct. This step would be used to address a problem with condenser air in-leakage. Ensuring steam seal pressure between 2 to 6 psig is step 6.2 of AOP 2574 and is under the Condenser Air In-leakage section. An in-leakage step will not improve vacuum if the loss is a result of a problem with heat transfer.
PLAUSIBLE: Ensuring steam seal pressure between 2 to 6 psig is a step in AOP 2574. If the problem is condenser air in-leakage due to low steam seal pressure then this would improve vacuum once the pressure is re-established between 2 - 6 psig.

B - INCORRECT: Throttle open the turbine exhaust hood spray bypass to lower exhaust hood temperatures is not correct. This step (6.4.1) is under the condenser air in-leakage section. This spray flow will not improve vacuum. The purpose of the hood sprays is to cool the turbine last stage blades and casing in the area of the last stage blades.
PLAUSIBLE: The examinee could reason, that like the pressurizer, spray will lower pressure in a saturated system, thus improving vacuum.

C - INCORRECT: Verify the operating SJAE steam supply pressure is 200 - 220 psig is not correct. This step would be used to address a problem with condenser air removal alignment. Verifying the operating SJAE steam supply pressure is 200 - 220 psig is step 4.1.c of AOP 2574 and is under the Condenser Air Removal Alignment section. An air removal alignment step will not improve vacuum if the loss is a result of a problem with heat transfer.
PLAUSIBLE: Verify the operating SJAE steam supply pressure is 200 - 220 psig is a step in AOP 2574. If the problem is with condenser air removal due to an improper SJAE steam supply pressure then this would improve vacuum once the pressure is re-established between 200 - 220 psig.

D - CORRECT: Refer to AOP 2517, Circulating Water Malfunctions, and raise circulating water pump speed is correct. Section 5 of AOP 2574 addresses a loss of condenser vacuum due to condenser heat transfer. The contingency actions (step 5.1.1.c) directs referring to AOP 2517, Circulating Water Malfunctions. Step 2.c response not obtained raises circ water pump speed to 100%. This will attempt to restore circulating water system flow to raise heat transfer.

References

AOP 2574 Loss of Condenser Vacuum, R008-00
AOP 2517 Circulating Water Malfunctions, R006
Main Turbine Lesson Text MT-00-C

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 051 000051 (APE 51) Loss of Condenser Vacuum / 4

Generic K/A Selected

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

Number 2.1.28 **RO** 4.1 **SRO** 4.1 **CFR Link** (CFR: 41.7)

G2.1.28 Knowledge of the purpose and function of major system components and controls.

Question #: **23** **Question ID:** **53445** Rev. 1 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/17/2018 11:19:07 AM ☑ RO ☐ SRO **Origin:** **Bank** ☐ Past NRC Exam?

A fuel handling accident occurs in the Spent Fuel Pool (SFP) area and an AEAS signal is generated. Select the statement which correctly describes the status of ventilation systems in the Spent Fuel Pool area.

☐ **A** Taking suction on outside air, discharging to the Enclosure Building Filtration System (EBFS).

☐ **B** Outside air supply isolated, discharging to the Main Exhaust System.

☐ **C** Taking suction on outside air, discharging to the Main Exhaust System.

☑ **D** Outside air supply isolated, discharging to the Enclosure Building Filtration System (EBFS).

Question Misc. Info: MP2*LORT*3128 [088 BPV-01-C 2757] (10/21/97) 2314F, AEAS, SFP, 2384

Justification

This question requires an understanding of the alignment of ventilation system that occur as a result of an Auxiliary Exhaust Actuation Signal (AEAS). The examinee must select the alignment the system will be in following an AEAS actuation. With an AEAS actuation present, and without a concurrent Enclosure Building Filtration Actuation Signal (EBFAS), the spent fuel pool will isolate for other areas and it's atmosphere will be discharged through the Enclosure Building Filtration system (EBFS) and out the Millstone Stack.

A - INCORRECT: Taking suction on outside air, discharging to the Enclosure Building Filtration System (EBFS) is not correct. The outside air supply is isolated when the AEAS signal stops supply fan F-20 and closes it's discharge damper HV-165. The supply of air is stopped so that EBFAS can draw a negative in the SFP area so that there will be no leakage from the area. Discharging to the Enclosure Building Filtration System (EBFS) is correct.
PLAUSIBLE: The normal alignment to the system has fresh air being drawn into the SFP by the operation of supply fan F-20. It is reasonable that you would want to bring in some fresh air in as long as any radiation released from the Fuel Handling Accident is filtered with EBFAS charcoal filters prior to being discharged to the outside atmosphere.

B - INCORRECT: Outside air supply isolated, discharging to the Main Exhaust System in not correct. Outside air supply isolated is correct but discharging to the Main Exhaust System in not correct. Both the normal suction and discharge are isolated. EBFS then draws a suction on the SFP and then discharges through charcoal filters to the monitored Millstone Stack.
PLAUSIBLE: Outside air supply isolated and discharging to the Main Exhaust System would create a greater negative in the SFP area. This would prevent the SFP atmosphere from leaking into adjacent areas. This is reasonable since the Auxiliary Building ventilation is designed to operate to create a slight negative in the Auxiliary Building. This prevents potentially radioactive areas from being discharge through an unmonitored path. Discharges through the Main Exhaust system ventilation are monitored but not filtered by charcoal. Creating a greater negative is a reasonable answer to assist in preventing an unmonitored radiation release from the SFP.

C - INCORRECT: Taking suction on outside air, discharging to the Main Exhaust System is not correct. Both the normal suction and discharge are isolated.
PLAUSIBLE: Taking suction on outside air, discharging to the Main Exhaust System is the normal system alignment if an AEAS is not present. It is reasonable for the examinee to think that the normal ventilation remains in service and the EBFAS system starts and takes a suction of the SFP area also to clean it up with charcoal.

D - CORRECT: Outside air supply isolated, discharging to the Enclosure Building Filtration System (EBFS) is correct. The SFP area supply fan F-20 is stopped and it's discharge damper closes. The normal discharge path to the main exhaust system is isolated. The discharge from the SFP area is aligned to EBFAS and discharged to the Millstone Stack.

References

Lesson Text RWV-00-C Radioactive Discharge Ventilation, R9
Lesson Text CCS-00-C Containment Systems, R10-03
AOP 2577 Fuel Handling Accident, R009
Fuel Handling Vent training Dwg

Question #: **23** **Question ID:** **53445** **Rev.** 1 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/17/2018 11:19:07 AM ☑ **RO** ☐ **SRO** **Origin:** **Bank** ☐ Past NRC Exam?

EBFAS AEAS Training Dwg

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 061 000061 (APE 61) Area Radiation Monitoring System Alarms / 7

Number AA1.01 **RO** 3.6 **SRO** 3.6 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

AA1.01 Ability to operate and / or monitor the following as they apply to the Area Radiation Monitoring (ARM)System Alarms: Automatic actuation

Question #: **24** **Question ID: 1100042** Rev. 3 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/17/2018 11:20:04 AM ☑ RO ☐ SRO **Origin: Bank** ☐ Past NRC Exam?

A Reactor trip occurred from 100% when the Main Turbine tripped. The following conditions exist:

- #1 PORV, RC-402 is stuck partially open (dual indication).
- Bus 24C is faulted.
- Pressurizer pressure is 1300 psia and stable.
- Average CET temperature is 577 °F and raising.
- RWST level is 94% and stable.
- Pressurizer level is 80% and rising.
- Reactor Vessel Level on both channels (RVLMS) is 80%.
- Containment pressure is 1.6 psig and slowly rising.
- EOP 2525, Standard Post Trip Actions have been completed.
- The crew has transitioned to EOP 2532, Loss Of Coolant Accident.

What action must be taken to ensure adequate core cooling?

☐ **A** Reduce RCS pressure using auxiliary spray to raise safety injection flow.

☑ **B** Perform a Controlled Cooldown using the condenser steam dump valves.

☐ **C** Power bus 24E from Unit 3 and start the "B" HPSI pump.

☐ **D** Eliminate the head void by starting all available CEDM cooling fans.

Question Misc. Info: MP2*LOIT, ICCS, CET, SCM, 2387, MB-05109, NRC-2011, 55.43(b)(5) [Vision# 386497]

Justification

A - INCORRECT: Reduce RCS pressure using auxiliary spray to raise safety injection flow is not correct. With a partially open PORV (and no other indicated break) lowering Pressurizer pressure by spray flow would move more of the RCS inventory into the Pressurizer and out of the open PORV. With a SBLOCA, RCS pressure, when at saturation, would be a function of the hottest source. At this time, that would be the core. Pressurizer spray flow will have little if any affect on the RCS pressure. Lowering pressure in the Pressurizer would simply cause more steam generation in the vessel.
PLAUSIBLE: Lowering RCS pressure by spray flow is a directed action in EOP 2532 (step 18). It could increase Safety Injection flow but would also lower subcooling. Spray flow with a pressurizer steam space leak and high pressurizer level will have little affect at lowering RCS pressure. A slight decrease in pressurizer pressure could cause the bubble to expand in the head and lower pressure.

B - CORRECT: Perform a Controlled Cooldown using the main steam dump valves to the condenser is 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. In a SBLOCA (which this is) safety injection flow is insufficient to cool the core. Steam generators must provide some amount of cooling to maintain adequate core cooling.

C - INCORRECT: Power bus 24E from Unit 3 and start the "B" HPSI pump is not correct. An additional HPSI pump would not help because the pumps are in parallel and HPSI is not injecting at an RCS pressure of 1300 psia (above HPSI pump shutoff head). Starting an additional HPSI pump would not raise HPSI discharge pressure above the existing RCS pressure.
PLAUSIBLE: Restoring power to a dead vital bus and recovering SI pumps is directed by EOP 2532 (step 5.b.1) to help regain control of RCS inventory.

D - INCORRECT: Eliminate the head void by starting all available CEDM cooling fans is not correct. Cooling the head would cool the steam bubble but it would take a long time for this to occur. In addition to balance RBCCW header loads, all CEDM coolers have been aligned to only the Fac. 1 RBCCW header. Therefore, none of the CEDM coolers will have RBCCW flow due to the loss of Fac. 1 vital power.
PLAUSIBLE: The CEDM cooler when they have RBCCW operating will cool the head. That is their purpose. It is reasonable for the examinee to understand the head could be cooled in this way. EOP 2532 does direct (step 10.e) that up to two CEDM fans remain running.

References

EOP 2532, Millstone Unit 2 Loss of Coolant Accident
PPC ICC Level Display
EOP 2541, Appendix 2, Millstone Unit 2 Figures
OP 2330A-001, RBCCW System Alignment, Facility 1
CEN-152 LOCA Bases Document

NO Comments or Question Modification History at this time.

Question #: **24** | ***Question ID:*** ***1100042*** | **Rev.** **3** | ☐ **Student Handout?** | ☐ **Lower Order?**

Last Edited: 6/17/2018 11:20:04 AM | ☑ ***RO*** | ☐ ***SRO*** | ***Origin:*** ***Bank*** | ☐ **Past NRC Exam?**

NRC K/A System/E/A **System** 074 000074 (EPE 74; W E06 & E07) Inadequate Core Cooling / 4

Number EA2.05 **RO** 3.4 **SRO** 4.2 **CFR Link** (CFR 43.5 / 45.13)

EA2.05 Ability to determine or interpret the following as they apply to a Inadequate Core Cooling: Trends in water levels of PZR and makeup storage tank caused by various sized leaks in the RCS

Question #: **25** | ***Question ID*: 8680010** | **Rev.** 1 | ☐ **Student Handout?** | ☑ **Lower Order?**

Last Edited: 6/15/2018 1:24:55 PM | ☑ ***RO*** | ☐ ***SRO*** | ***Origin*: Bank** | ☐ **Past NRC Exam?**

The plant was manually tripped form 100% power due to a rupture of the "A" Main Steam Header in the containment. On the trip, VA-20 was lost due to a fault on the bus. All other plant equipment is operating normally.

Where must the **#2 ADV** be operated from in order to stabilize RCS temperature?

☐ **A** The ADV controller on C21.

☑ **B** Local-Manual at the ADV.

☐ **C** The ADV controller on C05.

☐ **D** The ADV controller on C10.

Question Misc. Info: MP2*LOIT ESD, VA-20, 2504D, NRC-2008

Justification

A - WRONG; there are six different control power supplies to the two Atmospheric Dump Valves (ADV), three per ADV. The loss of each of these control power supplies causes a different impact on their applicable valve. The loss of VR-21 requires the #2 ADV be operated from C-21, however, the loss of VA-20 also de-energizes the C21 part of the ADV control circuit.
PLAUSIBLE; Student may confuse the different power supply loss effects.

B - CORRECT; VA-20 powers the entire #2 ADV control circuit outside of the control room. With a loss of VA-20, the #2 ADV can NOT be operated remotely from ANY location. The valve can be operated locally due to the location of the steam rupture (inside CTMT).

C - WRONG; The #2 ADV does not have the capability to be operated from C-05 upon the loss of VA-20.
PLAUSIBLE; Student may confuse the different power supply loss effects because the #1 ADV may be operated from its C05 controller upon a loss of one of its control power supplies.

D - WRONG; The C10 Fire Shutdown panel is designed for use when the control room must be evacuated, due to an Appendix "R" type fire. Although it is very protected due to its function, the loss of VA-20 will still prevent the operation of the #2 ADV from C10.
PLAUSIBLE; The student may assume that, based on the importance of C10 in plant safety and the protected design of its control circuitry, that even with the loss of VA-20 this panel would still be available.

References

AOP 2504D, R5C0, Pg. 3

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** A11 (CE A11**; W E08) RCS Overcooling Pressurized Thermal Shock / 4

Number AK2.1 **RO** 3.2 **SRO** 3.4 **CFR Link** (CFR: 41.7 / 45.7)

AK2.1 Knowledge of the interrelations between the (RCS Overcooling) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question #: **26** **Question ID: 2018019** Rev. 1 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/17/2018 11:20:54 AM ☑ **RO** ☐ **SRO** **Origin: Bank** ☐ Past NRC Exam?

The plant has experienced both a steam generator tube rupture (SGTR) and a small break loss of coolant accident (SBLOCA).

- Standard Post Trip Actions have been completed.
- The Functional Recovery procedure has been entered.
- Containment Integrity has been determined to be the first priority.
- Containment pressure has peaked at 3.2 psig and is slowly lowering.
- 2 RCPs are operating with adequate NPSH.
- All other plant equipment is operating as designed.

Cool down of the plant for isolation of the effected Steam Generator is the next step to be performed.

How will the plant be cooled down, and why is the plant cooled down to a Th of less than 515 °F.

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☐ **A** "A" Steam Dump valve to 50% open in manual, to prevent the opening of the Atmospheric Dump Valves, if a loss of offsite power were to then occur.

☐ **B** Atmospheric Dump Valves slowly in manual, to prevent the opening of the Atmospheric Dump Valves, if a loss of offsite power were to then occur.

☐ **C** Atmospheric Dump Valves to 50% open in manual, to prevent the opening of the Main Steam Safety valves, if a loss of offsite power were to then occur.

☑ **D** "A" Steam Dump valve slowly in manual, to prevent the opening of the Main Steam Safety valves, if a loss of offsite power were to then occur.

Question Misc. Info: New Question for 2018 ILT exam.

Justification

A - INCORRECT: "A" Steam Dump valve to 50% open in manual, to prevent the opening of the Atmospheric Dump Valve is not correct. Opening the "A" Steam Dump valve to 50% in manual is not correct because RCP are still operating. To prevent the opening of the Atmospheric Dump Valve on a loss of offsite power is also not correct. This is not a concern because the Atmospheric Dump Valve can be isolated.
PLAUSIBLE: Opening the "A" Steam Dump valve to 50% in manual is correct when RCPs are not operating. Cooling down to < 515 °F will also prevent the Atmospheric Dump Valves from opening.

B - INCORRECT: Atmospheric Dump Valve slowly in manual, to prevent the opening of the Atmospheric Dump Valve is not correct. Opening the Atmospheric Dump Valve slowly in manual is not correct since it is not the preferred path since the condenser is available. To prevent the opening of the Atmospheric Dump Valve on a loss of offsite power is also not correct. This is not a concern because the Atmospheric Dump Valve can be isolated.
PLAUSIBLE: Opening the Atmospheric Dump Valve slowly in manual would be correct if a Main Steam Isolation (MSI) signal occurred due to containment pressure rising to >/= 4.42 psig. If a MSI occurred the "A" Steam Dump would not be available because the condenser is not available. Procedure usage would direct the response not obtained step to be used, which is use of the Atmospheric Dump Valves. Cooling down to < 515 °F will also prevent the Atmospheric Dump Valves from opening.

C- INCORRECT: Atmospheric Dump Valve to 50% open in manual, to prevent the opening of the Main Steam Safety valves is not correct. Opening the Atmospheric Dump Valve to 50% in manual is not correct because the RCPs are still operating and it is not the preferred path since the condenser is available. To prevent the opening of the Main Steam Safety valves on a loss of offsite power is correct.
PLAUSIBLE: Opening the Atmospheric Dump Valve to 50% in manual would be correct if a Main Steam Isolation (MSI) signal occurred due to containment pressure rising to >/= 4.42 psig and RCPs were not operating. If a MSI occurred the "A" Steam Dump would not be available because the condenser is not available. Procedure usage would direct the response not obtained step to be used, which is use of the Atmospheric Dump Valves.

D - CORRECT: The correct answer is opening the "A" steam dump slowly in manual until Th in both loops is less than 515 F. This prevents the opening of the Main Steam Safety valves on a loss of offsite power after the effected Steam Generator has been isolated. The examinee must recognize that a MSI has not been actuated (containment pressure is < 4.42 psig) so the condenser is available and the "A" steam dump valve can be used. The "A" steam dump valve is the expected response. Procedure usage rules has the expected response step performed if the equipment is available before the response not obtained step is used. If possible steaming should be to the condenser to prevent discharge of radioactivity to the environment. The functional recovery implementation guide specifies how much the Atmospheric Dump valve and "A" Steam Dump valve should be open depending on whether RCPs are operating or not. The valves are opened slowly when RCPs are operating in an attempt to maintain RCP NPSH. The RCS is cooled down to < 515 °F to prevent the main steam safety valves from lifting if offsite power is lost (RCP lose power and RCS temperature rises due to natural circulation).

Question #: **26** **Question ID: 2018019** Rev. 1 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/17/2018 11:20:54 AM ☑ **RO** ☐ **SRO** **Origin: Bank** ☐ Past NRC Exam?

References

CEN-152 Rev. 5.4, Steam Generator Tube Rupture Recovery Bases
OP 2260, EOP 2540, "Functional Recovery," Implementation Guide
EOP 2541, Appendix 12, Millstone Unit 2 SGTR Response
Safety Functional Requirements Manual 3.6.2 Steam Generator Tube Rupture
EOP 2540E, Millstone Unit 2 Functional Recovery of Containment Isolation

NO Comments or Question Modification History at this time.

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

Number EK3.3 **RO** 3.7 **SRO** 3.9 **CFR Link** (CFR: 41.5 / 41.10, 45.6, 45.13)

Knowledge of the reasons for the following responses as they apply to the (Functional Recovery) Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.

Question #: **27** **Question ID: 6800008** Rev. 2 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/14/2018 9:08:18 AM ☑ **RO** ☐ **SRO** **Origin: Bank** ☐ Past NRC Exam?

The plant was tripped from 100% power due to a stuck open pressurizer spray valve.

All four RCPs were secured, however the Steam Dump Tavg Controller, TIC-4165, <u>remains</u> in automatic control.

Which of the following describes the effect of having TIC-4165 in automatic control under these conditions?

☐ **A** All four steam dumps would immediately reopen, causing an RCS cooldown.

☐ **B** As Natural Circulation develops, RCS temperature will rise until it is stabilized by TIC-4165.

☑ **C** When RCS temperature rises to develop Natural Circulation, all four steam dumps will open causing an RCS cooldown.

☐ **D** When RCS temperature rises to develop Natural Circulation, both ADVs will open and stabilize RCS temperature.

Question Misc. Info: MP2*LOIT RRS-01-C, 2386, RRS, NC

Justification

A - WRONG; Although Foxboro has taken control, and Tavg (537°F) is presently above the auto setpoint (535°F) with a turbine trip signal present, the valves have already completed their post-trip response and closed. The valves will not reopen until Tavg exceeds 540°F.
PLAUSIBLE; The student may remember Foxboro shifts control to auto, but not know the follow-up response once the valves have closed.

B - WRONG; Because the Foxboro IA is programmed to always take control in auto and ensure the auto setpoint of TIC-4165 is at 535°F. However, unlike a normal proportional controller, it will NOT modulate the steam dumps open in response to input-setpoint mismatch until Tavg exceeds 540°F, where it will open <u>all four</u> steam dumps about 8% - 10%.
PLAUSIBLE; The student may remember that the Foxboro IA is designed as a proportional control system and will take control of TIC-4165 with a setpoint of 535°F; but believe Foxboro will then control temperature in auto as a normal proportional controller.

C - CORRECT; TIC-4165 setpoint is fixed at a Tavg of 535 °F, and on a plant trip would fully open all four condenser steam dumps to lower Tavg to a "no-load" value. At 535 °F, the controller output would be just closing the valves if temperature were dropping from the post-trip response. However, if Tavg begins to rise again once the valves have been closed, the control system is programmed to NOT reopen the valves until Tavg gets to 540 °F. At that Tavg, the valves would get about a 25% demand signal, which equates to about an 8% - 10% open position on <u>all four</u> steam dumps. Because post-trip decay heat loads requires only one steam dump to open about 25%, suddenly opening all four steam dumps 8% - 10% would cause an excessive cooldown. Because of this "expected" effect of the steam dump control system design, if RCS forced circulation is lost on a plant trip, the RO is required to place TIC-4165 in Manual with a zero output, thus preventing the steam dumps from opening as RCS temperature rises above 540 °F with the development of natural circulation. When the Foxboro IA system takes control of TIC4165 controller, it always takes control in AUTO with the setpoint at the programmed value of 535 °F and all auto response functions restored. Therefore, in the scenario given, as Thot rises and drives Tavg above 540 °F, Foxboro's control of TIC-4165 will open all four condenser steam dumps about 8% and cause an immediate cooldown.

D - WRONG; Although the Foxboro IA will always take control in auto and shift the setpoint to a value that would ensure the steam dumps go closed, in the case of TIC-4165 it relies on the lack of a turbine trip signal to drive the controller output to zero. That trip signal is now present, so TIC-4165 will respond according to input-setpoint mismatch based on its programming.
PLAUSIBLE; Student may remember that Foxboro tracks the C05 controller and takes control in auto, therefore it should respond the same when control is transferred from C05 to Foxboro.

K/A Match; The question requires knowledge of an unusual plant system/component operating characteristic that is unique to RCS temperature response during the development of Natural Circulation.

References

EOP 2525, SPTA Tech Guide, R27, Pg 19

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** A13 (CE A13) Natural Circulation Operations

Number AK3.1 **RO** 3.4 **SRO** 3.7 **CFR Link** (CFR: 41.5 / 41.10, 45.6, 45.13

AK3.1 Knowledge of the reasons for the following responses as they apply to the (Natural Circulation Operations): Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.

Question #: **28** **Question ID: 53695** **Rev. 1** ☐ Student Handout? ☑ Lower Order?

Last Edited: 5/30/2018 4:18:48 PM ☑ **RO** ☐ **SRO** **Origin: Bank** ☐ Past NRC Exam?

A plant startup is in progress with preparations being made to start the first RCP.

Which one of following annunciators being in alarm could indicate the presence of an interlock which will PREVENT the starting of the "A" RCP?

☐ **A** RCP A CLG WTR TEMP HI

☐ **B** RCP A LWR OIL RSVR LEVEL LO

☑ **C** RCP A CLG WTR FLOW LO

☐ **D** RCP A BLEED OFF FLOW HI

Question Misc. Info: MP2*LORT*4674 (RCS-01-C 4932), 2301C, RCP, RCS

Justification

A - WRONG; Although this would "administratively" prevent the starting of the RCP, there is no interlock function driven by this RCP parameter.
PLAUSIBLE; Student may recall that this alarms requires immediate tripping of the RCP and, therefore, falsely assume it must drive an interlock to prevent starting it.

B - WRONG; Although this may "administratively" prevent the starting of the RCP, there is no interlock function driven by this RCP parameter.
PLAUSIBLE; The RCP Lift Pump must be running with good discharge pressure to satisfy the other RCP start interlock. Student may confuse the administrative oil "level" requirement with the interlock oil "pressure" requirement.

C - CORRECT; ARP 2590B-071 (C03; DA-17) states:
4. IF preparing to start RCP and alarm does not clear when breaker is racked in, PERFORM the following:
   4.1 SEND operator to adjust cooling water flow (enables RCP cooling water flow start permissive).
   4.2 IF cause of the alarm has been determined and corrected, Refer To OP 2301C, "Reactor Coolant Pump Operation" and START "A" RCP as required.

D - WRONG; Although this may "administratively" prevent the starting of the RCP, there is no interlock function driven by this RCP parameter.
PLAUSIBLE; Per ARP 2590B, this could be indication of a Vapor Seal failure, which requires an immediate plant trip. Student might conclude that an alarm potentially requiring an immediate trip of the plant and RCP would also have an interlock to prevent the pump from being started.

References

ARP 2590B-071, R0C0

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 003 003 (SF4P RCP) Reactor Coolant Pump

Number A4.06 **RO** 2.9* **SRO** 2.9 **CFR Link** (CFR: 41.7 / 45.5 to 45.8)

A4.06 Ability to manually operate and/or monitor in the control room: RCP parameters

Question #: **29** **Question ID: 2018011** Rev. 0 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/1/2018 3:56:17 PM ☑ RO ☐ SRO **Origin: New** ☐ Past NRC Exam?

The plant is in Mode 1 when “A” RCP bleedoff flow suddenly spikes to over 20 gpm due to a degraded seal.

Which of the following plant system design features would prevent this excessive RCS seal bleedoff flow from affecting the other RCPs?

☐ **A** RCP Bleedoff Flow Control Valve will throttle to control bleedoff flow and pressure within normal limits.

☑ **B** “A” RCP Bleedoff Excess Flow Check Valve will close and isolate bleedoff flow from the seal package.

☐ **C** A high bleedoff flow annunciator will alert the crew to manually divert “A” RCP bleedoff flow to the PDT.

☐ **D** "A" RCP bleedoff flow isolation valves will automatically divert bleedoff flow from the VCT to the PDT.

Question Misc. Info: MP2*LOIT RCP, Bleedoff, Seals

Justification

A - WRONG; The bleedoff flow control valve throttles to maintain bleedoff system pressure based on the flow coming from the seals. If flow from one RCP became excessive, it would OPEN to ensure general bleedoff pressure remained at the setpoint value.
PLAUSIBLE; Student may recognize the above system reaction, but fail to consider that the "A" RCP bleedoff will now be at a much higher pressure than the other three RCPs. Therefore, bleedoff flow from the "A" RCP would block bleedoff flow from the other RCPs if the check valve didn't close.

B - CORRECT; Each RCP has an Excess Flow Check valve in it’s bleedoff flow path. This check valve is designed to close if bleedoff flow exceeds 10 gpm, and stop the excessive flow from affecting the other RCPs. When the bleedoff flow stops, pressure across the affected RCP's seal will equalize with primary pressure and differential pressure across each seal stage will go to zero.

C - WRONG; This is only true if the normal bleedoff flow path out of CTMT is isolated (by something like a CIAS). This would occur if the check valve did NOT seat and the bleedoff flow exceeded the flow capacity of the system relief valve.
PLAUSIBLE; Student may recognize the excess flow check valve will bottle up the "A" RCP seal leakoff, but not remember the exact flowpath and believe diverting flow to the PDT would give the bottled up bleedoff flow another release path.

D - WRONG; A possible scenario if when the bleedoff check valve closed, flow was diverted to the PDT by way of the bleedoff line relief valve. However, although this relief valve may possibly lift on the initial high bleedoff flow "spike", it would be isolated from the affected RCP when the high flow check valve closes.
PLAUSIBLE; Student may remember seal flow/pressure would cause the relief to lift when seal flow is isolated by closing the CTMT system isolation valve (like a CIAS) and consider that the expected outcome if an excess flow check valve closes.

References

ARP 2590B-068, R1C0

NO Comments or Question Modification History at this time.

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

Number K4.07 **RO** 3.2 **SRO** 3.4 **CFR Link** (CFR: 41.7)

K4.07 Knowledge of RCPS design feature(s) and/or interlock(s) which provide for the following: Minimizing RCS leakage (mechanical seals)

Question #: **30** **Question ID:** **75164** **Rev.** 4 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/17/2018 11:21:53 AM ☑ **RO** ☐ **SRO** **Origin:** **Bank** ☐ Past NRC Exam?

The plant is in normal operation at 100% power, with the following CVCS conditions:

- "A" Charging Pump is aligned as the "first backup" pump.
- "B" Charging Pump is in "Pull-To-Lock".
- "C" Charging Pump is aligned as the "running" pump and is operating normally.

The RO has manually started the "A" charging pump and adjusted the bias thumbwheel on HIC-110 (letdown flow controller) to stabilize pressurizer level at setpoint.

Five minutes later, the "A" charging pump trips on overload (breaker fault). With no operator action. Which of the following is the expected plant response after the "A" charging pump has tripped?

☐ **A** Letdown will isolate on high temperature and Pressurizer level will rise with no signal available to stop the "C" Charging pump.

☐ **B** Pressurizer level control will lower letdown flow and maintain pressurizer level at setpoint with only the "C" charging pump running.

☐ **C** HIC-110 bias will maintain letdown flow constant and cause pressurizer level to continue to lower until the pressurizer is empty.

☑ **D** Pressurizer level control will lower letdown flow and stabilize pressurizer level below setpoint with only the "C" charging pump running.

Question Misc. Info: MP2*LOIT PLC-01-C, 2304A, PLPCS, NRC, APP

Justification

A - INCORRECT: Letdown will isolate on high temperature and Pressurizer level will rise with no signal available to stop the "C" Charging pump is not correct. Letdown will not isolate because you still have charging flow through the regenerative heat exchanger.
PLAUSIBLE: Letdown does isolate on high temperature if only one charging pump is running and it is lost. The examinee could reason that something similar could occur if two charging pumps are operating and one is lost.

B - INCORRECT: Pressurizer level control will lower letdown flow and maintain pressurizer level at setpoint with only the "C" charging pump running is not correct. A proportional controller will not restore level back to it's original value.
PLAUSIBLE: The examinee may think that the level controller is a proportional controller with integral action which would restore the system back to setpoint. This would be supported by the fact that pressurizer level is normally at setpoint.

C - INCORRECT: HIC-110 bias will maintain letdown flow constant and cause pressurizer level to continue to lower until the pressurizer is empty is not correct. The pressurizer level control will still operate to lower letdown to stabilize pressurizer level prior to the pressurizer emptying,
PLAUSIBLE: The examinee could reasonable believe that since the bias was manually adjusted to a new higher letdown flow that it fixed at this value and it will not change. Therefore letdown will stay at the higher value.

D - CORRECT: Pressurizer level control will lower letdown flow and stabilize pressurizer level below setpoint with only the "C" charging pump running is correct. With letdown set, by the letdown flow control bias for two charging pump operation, it's flow will be ~ 84 gpm. Once a charging pump is lost pressurizer level will start to lower since charging will be at 44 gpm and letdown will be at 84 gpm. The pressurizer level controller is a proportional controller. There must be a level deviation for the system to respond. As pressurizer level lowers the control system will lower letdown flow. Once pressurizer level lowers enough, letdown will lower to a point that stabilizes pressurizer level, at a point below programmed level. A proportional controller will not restore level back to it's original value. Only the "C" charging pump is operating because "A" tripped and "B" charging pump will not start on any signal when in "Pull-To-Lock".

References

Lesson Text PLC-01-C, Pressurizer Level & Pressure Control System, R6

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 004 004 (SF1; SF2 CVCS) Chemical and Volume Control

Number K3.07 **RO** 3.8 **SRO** 4.1 **CFR Link** (CFR: 41.7/45/6)

K3.07 Knowledge of the effect that a loss or malfunction of the CVCS will have on the following: PZR level and pressure

Question #: **31** **Question ID:** **55218** **Rev.** 7 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/14/2018 9:08:46 AM ☑ **RO** ☐ **SRO** **Origin:** **Bank** ☐ Past NRC Exam?

The plant is shutdown, in Mode 5, with Shutdown Cooling (SDC) in service.
The RCS level is being maintained at the centerline of the hot leg.

What is the highest (maximum) SDC flow allowed during the above conditions?

☐ **A** 1400 gpm

☑ **B** 1600 gpm

☐ **C** 4000 gpm

☐ **D** 4800 gpm

Question Misc. Info: MP2*LOIT*2540 [005 SIP-01-C 7386] (8/16/96) 2310, 2301E, SDC, APP

Justification

A - WRONG: This is the MINIMUM limit for SDC flow while in RIO, not the MAXIMUM. This minimum flow requirement ensures SDC flow is maintained above the Tech. Spec. Limit of 1000 gpm, even with flow measurement inaccuracies.
PLAUSIBLE; Student may remember the Tech. Spec. limit for SDC flow and that this is also a limit, but not which limit.

B - CORRECT: OP 2301E, *Draining The RCS*, Caution states "Due to vortexing at SDC suction line in hot leg when drained to centerline, SDC flow must be limited to between 1400 and 1600 gpm."
Also, OP-2310, *Shutdown Cooling System*, Discussion section states the same limit for SDC flow during Reduced Inventory Conditions.

C - WRONG: This is the limit for SDC flow through the (tube side) SDC heat exchangers.
PLAUSIBLE; Student may recall that this **is** a limit for SDC flow, but not which one.

D - WRONG: This is the maximum flow limit for RBCCW through the (shell side) SDC HXs.
PLAUSIBLE; Student may recall this as a flow rate limit for the system, but not which one.

References

OP 2310, Shutdown Cooling, R32C0, Pg 6

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 005 005 (SF4P RHR) Residual Heat Removal

Number A4.03 **RO** 2.8* **SRO** 2.7* **CFR Link** (CFR: 41.7 / 45.5 to 45.8)

A4.03 Ability to manually operate and/or monitor in the control room: RHR temperature, PZR heaters and flow, and nitrogen

Question #: **32** **Question ID: 2018012** Rev. 0 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/25/2018 1:34:48 PM ☑ RO ☐ SRO Origin: New ☐ Past NRC Exam?

The plant experienced a main steam line rupture on steam generator #2. The crew has transitioned to EOP 2536, Excess Steam Demand event. The affected Steam Generator has boiled dry and RCS temperatures have been stabilized. The RO has been directed to throttle or secure HPSI injection flow and control charging and letdown, in accordance with the EOP. The following conditions exist:

- Reactor power is 1 X 10-5% and lowering
- Subcooling is 160 °F and rising
- Pressurizer Pressure is 1325 psia and rising
- Pressurizer level is 72% and rising
- Steam Generator #1 level is 33% and rising being fed from Auxiliary Feedwater
- Reactor vessel level is 61% and stable

Which of the following pumps should be secured and why?

☐ **A** Both running HPSI pumps should be secured to stabilize pressurizer level.

☐ **B** Throttle HPSI flow to maintain Emergency Boration and lower the rate of rise in the pressurizer.

☐ **C** HPSI Throttle/Stop criteria is NOT met and therefore full HPSI and charging flow must be maintained.

☑ **D** All running HPSI and charging pumps should be secured to stabilize pressurizer level.

Question Misc. Info: MP2*ILT ESD, 2536, SI, throttle

Justification

A - INCORRECT: Both running HPSI pumps should be secured to stabilize pressurizer level is not correct. This is not correct because stopping HPSI pumps will not stop the injection to the RCS. Once HPSI pumps are secured pressurizer level will continue to rise at the same rate. HPSI is not injecting because RCS pressure is above the HPSI pump discharge pressure.
PLAUSIBLE: The step to stop HPSI is the HPSI Throttle/Stop Criteria step in the EOP. It can significantly lower flow to the RCS and the refill of the pressurizer if RCS pressure is less than the HPSI pump discharge pressure. And in a design bases Excess Steam Demand event HPSI will inject for a portion of the event until temperature is stabilized and inventory is recovering.

B - INCORRECT: Throttle HPSI flow to maintain Emergency Boration and lower the rate of rise in the pressurizer is not correct. OP 2260 states that all safety injection can be stopped once HPSI throttle stop criteria is met if pressurizer level is approaching the upper end of the control band and reactor power is stable or dropping. Therefore a minimum of 40 gpm of either charging or HPSI is not required.
PLAUSIBLE: Emergency Boration at >/= 40 gpm is required if a Safety Injection Actuation Signal (SIAS) is present and the SIAS signal has not been reset. Resetting the SIAS would not be possible at this time because containment pressure would be above 4.42 psig.

C - INCORRECT: HPSI Throttle/Stop criteria is NOT met and therefore full HPSI and charging flow must be maintained. This is not correct. HPSI Throttle/Stop criteria is met.
PLAUSIBLE: The examinee may not remember the criteria

D - CORRECT: All running HPSI and charging pumps should be secured to stabilize pressurizer level. Secure HPSI pumps because they are no longer required. Secure charging pumps, because at > 1250 psia in the RCS, HPSI pumps are not injecting but charging pumps are. Charging pumps are filling the RCS and causing pressurizer level to rise. All injection can be secured if HPSI throttle stop criteria is met, pressurizer level is approaching the upper end of the control band, and reactor power is stable or dropping (OP 2260). The conditions provided in the stem meet this criteria. If pressurizer level is allowed to fill the RCS solid, a high pressure condition will exist and Pressurized Thermal Shock (PTS) could occur.

References

EOP 2536, Excess Steam Demand R027-00
OP 2260 EOP Users Guide R015-00
ICC Lesson Plan Figure
CEN-152 R6

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 006 006 (SF2; SF3 ECCS) Emergency Core Cooling

Question #: **32** **Question ID: 2018012** **Rev. 0** ☐ **Student Handout?** ☐ **Lower Order?**

Last Edited: 6/25/2018 1:34:48 PM ☑ **RO** ☐ **SRO** **Origin: New** ☐ **Past NRC Exam?**

Number A1.07 **RO** 3.3 **SRO** 3.6 **CFR Link** (CFR: 41.5/45.5)

A1.07 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Pressure, high and low

Question #: **33** **Question ID:** **8054464** **Rev.** 2 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/18/2018 2:02:44 PM ☑ **RO** ☐ **SRO** **Origin:** **Bank** ☑ **Past NRC Exam?**

The plant is at 100% power, steady state, with all equipment operating as designed.
Then, an RCS Safety Valve begins leaking by, causing a slow rise in Quench Tank parameters.

IAW OP 2301A, "PDT and Quench Tank Operation", which of the following is required to ensure the Quench Tank will be maintained within its design limits?

☑ **A** Quench Tank cooling must be manually initiated, as required.

☐ **B** The Quench Tank must be aligned to continuously drain to the PDT.

☐ **C** The Quench Tank pressure must be continuously vented.

☐ **D** Quench Tank gas space must be regularly sampled for hydrogen concentration.

Question Misc. Info: MP2*LOIT*0634 [002 RCS-01-C 4928] (8/15/96) 2301, RCS, NRC-2016

Justification

A - CORRECT; Quench Tank cooling is NOT normally aligned to the tank and must be manually initiated when required. If this is not done, the tank could over-pressurize (blow out rupture disk) and the water could boil off. Too low a water level would prevent the tank from performing as designed.

B - WRONG; There is NO automatic level control valve, or drain piping geometry, that will stop the Quench Tank from completely emptying into the PDT once it is aligned to drain there. Therefore, when the Quench Tank is aligned to drain to the PDT, the dropping level must be closely monitored and the drain valve closed once the proper level is reached.
PLAUSIBLE; Examinee may confuse the existence of the QT/PDT recirc pump with the PDT transfer pumps. The recirc pump is used to cool either tank or refill their reference legs, but is not needed to move water between them.

C - WRONG; The pressure regulator for the Quench Tank does not function in an automatic mode, and leaving it open could result in the generation of excessive amounts of gaseous rad waste.
PLAUSIBLE; Examinee may believe the pressure regulator is designed to function just like any other pressure regulator of similar construction, but the QT regulator has never worked in auto mode.

D - WRONG; The Quench Tank gas space is expected to contain a high concentration of hydrogen, because as water from an RCS Safety or PORV enters the tank, it will depressurizes and the entrained gasses will come out of solution. Even though the QT has a rupture disc designed to vent the tank to CTMT before pressure exceeds design limits, there is no administrative requirement to continuously sample the tank gas space for hydrogen.
PLAUSIBLE; Examinee may recall that the main control board only displays a nitrogen supply to the QT and PDT, not hydrogen. Therefore, if only nitrogen is added, it may be logically deduced that excess hydrogen in the QT could cause the amount of hydrogen released into CTMT during an event to alter that assumed in the Analysis.

References

OP 2301A, PDT and Quench Tank Operation, R16C0, St. 4.1, "Recirculation and Cooling of QT and PDT"

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 007 007 (SF5 PRTS) Pressurizer Relief/Quench Tank

Number A1.03 **RO** 2.6 **SRO** 2.7 **CFR Link** (CFR: 41.5/45.5)

A1.03 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including: Monitoring quench tank temperature

Question #: **34** **Question ID:** **53297** **Rev.** 3 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/14/2018 9:09:44 AM ☑ **RO** ☐ **SRO** **Origin:** **Bank** ☐ Past NRC Exam?

A midcycle reactor start up is in progress using OP 2202, Reactor Startup ICCE.
An ECP has been performed for Group 7 @ 30 steps.

The RO stops CEA withdrawal with Group 4 at 65 steps due to a count rate doubling.
Reactor power is continuing to rise with a stable positive startup rate.

Then, while awaiting the completion of readings, numerous alarms on C-04 alert the crew that the Level 1 and Level 2 wide range bistables have just lit (armed) on all 4 RPS channels.

Based on this information, which one of the following is the required course of action?

☑ **A** Immediately fully insert all regulating CEAs.

☐ **B** Have R.E. perform a 1/M plot before moving CEAs further.

☐ **C** Verify the wide range meters on C-04 shift from "CPS" to "%".

☐ **D** Stabilize at 10E-4% and take critical readings.

Question Misc. Info: MP2*LORT*1954 [015 NIS-01-C 4713] (1/9/97) 2202, ECP, SDM, CEA

Justification

A - CORRECT; The reactor is normally critical between 10-5% and 10-4% power. RG-4 must be >/= 72 steps withdrawn before the reactor is critical or the CEAs will not be above the Tech. Spec. Transient Insertion Limit. The Lvl 1/2 bistables arming indicate the reactor is at ~ 10-4% power, where the CEDS PDIL interlock arms to ensure the CEAs are withdrawn to Gp-4 >/= 72 steps. Therefore, this indicates the reactor is going critical to early and the reactor startup procedure requires that the regulating CEAs be inserted and a new ECP be calculated.

B - WRONG; Reactor power is at ~ 10-4%, which is too high with the CEAs below the Transient Insertion Limit.
PLAUSIBLE; Student may realize that a 1/M plot would "reset" the estimate as to what rod height RE predicts criticality to occur, which would be a check as to whether a reactivity anomaly exists. The procedure directs this action be performed when a count rate doubling occurs.

C - WRONG; Although this may occur at this power level (10-4%), power is too high for GP-4 CEAs to be below 72 steps withdrawn.
PLAUSIBLE; Student may remember the procedure states this may occur at 10-4% power and directs that the automatic action be verified.

D - WRONG; The reactor may be critical at this point, but regardless, power is too high for CEAs to be inserted as far as stated.
PLAUSIBLE; Student may remember the procedure directs power be stabilized and readings taken when criticality is achieved, but forget it is stabilized at 10-3% for readings to be taken.

References

OP-2202, R25C0, Reactor Startup, Discussion section, NOTE for St. 4.3.8

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.2.1 **RO** 4.5 **SRO** 4.4 **CFR Link** (CFR: 41.5 / 41.10 / 43.5 / 45.1)

G2.2.1 Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

Question #: **35** **Question ID:** **78985** **Rev.** 5 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/15/2018 1:36:45 PM ☑ **RO** ☐ **SRO** **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 required status of the following two components?

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

☐ **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; which means the "A" EDG would power 24E (through 24C) on a LOOP. Therefore, the SIAS/LNP hand switch must be left in the Block position to prevent the EDG from powering both the normal facility ("A") RBCCW pump and the swing ("B") RBCCW pump. If Bus 24D was supplying Bus 24E, then the SIAS/LNP hand switch would be placed in the Normal position and the annunciator would reset (NOT in alarm). 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.

***K/A Match**; Knowing the normal and emergency power supplies to the "B" RBCCW Pump and the function of the "block" switch (on a LOOP) is key to determining the final status of the switch and annunciator. Note that at Millstone Unit 2, the Loss Of Offsite Power (LOOP) is usually referred to as a Loss of Normal Power (LNP). Also, the EDG are automatically started on a Safety Injection Actuation Signal (SIAS) to prewarm the diesel engines and help ensure their availability in case offsite power is lost during an accident. Hence the "SIAS/LNP" wording for the block switch.*

References

AOP 2564, Loss of RBCCW, R5C0, St 6.1h

NO Comments or Question Modification History at this time.

Question #: **35** **Question ID:** **78985** **Rev.** 5 ☐ **Student Handout?** ☐ **Lower Order?**

Last Edited: 6/15/2018 1:36:45 PM ☑ **RO** ☐ **SRO** **Origin:** **Bank** ☐ **Past NRC Exam?**

NRC K/A System/E/A **System** 008 008 (SF8 CCW) Component Cooling Water

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

K2.02 Knowledge of bus power supplies to the following: CCW pump, including emergency backup

Question #: **36** | **Question ID:** **55184** | **Rev.** 3 | ☐ Student Handout? | ☐ Lower Order?

Last Edited: 6/1/2018 4:49:51 PM | ☑ **RO** | ☐ **SRO** | **Origin:** **Bank** | ☐ Past NRC Exam?

The plant is at 100% power with Channel "Y" pressurizer pressure and level control selected as the controlling channels.

Then, a small leak develops in the pressurizer level Channel "Y" reference leg.

As the level in the reference leg slowly drops, how will the pressurizer heaters respond?

☐ **A** When the level in the leaking reference leg eventually reaches 20%, all backup heaters will trip and both proportional heater breakers will trip.

☐ **B** When the level in the leaking reference leg eventually reaches 20%, all backup heaters will trip and proportional heaters will be at maximum.

☑ **C** Channel "Y" level will slowly rise and when ~4% above setpoint, Channel "Y" level control will drive proportional heaters to maximum and turn on all backup heaters.

☐ **D** Channel "Y" level will slowly rise and when ~4% above setpoint, proportional heaters will go to maximum but Channel "Y" pressure control will keep backup heaters off.

Question Misc. Info: MP2*LOIT*2517 [011 PLC-01-C 4814] (8/16/96) 2304A, PLPCS, NRC, APP

Justification

A - WRONG; The PZR heater protective trip on low level is based on the output of the controlling channels DP cell, not the actual level of the reference leg.
PLAUSIBLE; If the affect of a leaking variable leg were confused with that of a leaking reference leg, then this would be the expected system response.

B - WRONG; The PZR heater protective trip on low level is based on the output of the controlling channels DP cell and it would trip ALL PZR heaters, even the proportional banks.
PLAUSIBLE; If the affect of a leak in the variable leg and reference leg were confused, and it was assumed that the Tech. Spec. required proportional heaters would not be subject to the failure of a Control Channel DP cell, then this would be the expected system response.

C - CORRECT; As the reference level drops, indicated level on the controlling Channel "Y" will slowly rise, eventually exceeding the "insurge" setpoint of +3.6%. This will turn on all banks of backup heaters and drive the proportional heaters to maximum output. Although real level will drop, all heaters will still be on until actual PZR level reaches <20% (as seen by the unaffected Channel "X").

D - WRONG; The loss of level in the reference leg of the controlling Channel "Y" will be seen as a rise in PZR level, causing Channel "Y" to increase letdown flow and lower actual PZR level.
PLAUSIBLE; The high PZR pressure signal that overrides ALL signals to turn on the backup heaters may be confused with what is causing the proportional heaters to go to maximum.

References

ARP 2590B-217, R2C0 (C-02/3, A-39)

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 010 010 (SF3 PZR PCS) Pressurizer Pressure Control

Number K6.02 **RO** 3.2 **SRO** 3.5 **CFR Link** (CFR: 41.7 / 45.7)

K6.02 Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: PZR

Question #: **37** **Question ID: 1100018** Rev. 1 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/15/2018 1:38:51 PM ☑ RO ☐ SRO **Origin: Bank** ☐ 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 other inputs to RPS are unchanged and all other CPC circuits are functioning as designed.

Which of the following would result from this malfunction?

☑ **A** The channel's DNB protection trip setpoint would calculate to a higher value.

☐ **B** Several trips would be triggered by a Power Trip Test Interlock (PTTI) actuation.

☐ **C** Trip calculations would now be from the channel's Delta-T Power.

☐ **D** The channel would trigger a CEA Withdrawal Prohibit actuation.

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

Justification

A - CORRECT; The RPS trip on TM/LP is designed to protect the core against DNB. 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.

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 circuitry.

C - 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.

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

LP RPS-01-C, R7C0, Pg 20-21

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 012 012 (SF7 RPS) Reactor Protection

Number K5.01 **RO** 3.3* **SRO** 3.8 **CFR Link** (CFR: 41.5 / 45.7)

K5.01 Knowledge of the operational implications of the following concepts as the apply to the RPS: DNB

Question #: **38** **Question ID: 2018014** **Rev. 0** ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/14/2018 9:10:24 AM ☑ **RO** ☐ **SRO** **Origin: Mod** ☐ Past NRC Exam?

The plant was manually tripped when a pressurizer spray valve failed open.
The following conditions exist:

- All RCPs were secured to stop the RCS depressurization.
- EOP 2528, Loss Of Offsite Power/Loss of Forced Circulation is in progress.
- A Natural Circulation cooldown to Mode 4 has been started.
- "B" Main Feed pump is supplying feed flow.
- The "A" Condenser steam dump valve is being used for the cooldown.
- Facility 1 MSI has just been manually blocked.
- Facility 2 MSI has NOT yet been blocked.

Then, a problem with #1 SG feed control causes a sudden excess feed flow to the SG.
Before the crew can stabilize the plant, SG pressures drop to the following values:

- #1 SG pressure drops to 568 psia.
- #2 SG pressure drops to 670 psia.

Which of the following describes the impact of the sudden SG pressure drop, and the actions taken to control RCS heat removal?

☐ **A** MSI actuation has closed the #2 MSIV only. Override open #2 MSIV, continue using Main Feed Water and the "A" condenser steam dump valve.

☑ **B** MSI actuation has closed both of the MSIVs. Verify both MSIVs closed, switch to Auxiliary Feed Water and use the atmospheric dump valves.

☐ **C** MSI actuation has closed the #2 MSIV only. Block and reset MSI, continue using Main Feed Water and the "A" condenser steam dump valve.

☐ **D** MSI actuation has closed both MSIVs. Block and reset MSI, open the MSIVs, switch to Auxiliary Feed Water and use the Atmospheric dump valves.

Question Misc. Info: MP2*LOIT ESA; NRC-2014

Justification

A - WRONG; The MSI block was activated for only facility 1 during the controlled cooldown. However, either SG dropping below the ESAS MSI setpoint of 572 psia will trigger an MSI actuation on Facility 2 because it is not yet blocked. Either Facility of MSI actuating will close BOTH MSIVs.
PLAUSIBLE; Student may recall MSI can still occur if one SG is isolated due to an event and the other continues to depressurize. However, they may feel because Facility 1 was already blocked, that Facility's MSIV should not get a close signal.

B - CORRECT; The MSI block was activated for Facility 1 ONLY during the controlled cooldown. As a result of the excess feed flow event, #1 SG pressure dropped below the ESAS MSI setpoint of 572 psia, which will trigger an MSI actuation on Facility 2 of ESAS because it is not yet blocked. Either ESAS Facility of MSI actuating will close BOTH MSIVs.

C - WRONG; Either Facility of MSI actuating will close BOTH MSIVs.
PLAUSIBLE; Student may confuse the action of ESAS on an MSI trigger of one Facility and be considering the actions taken if an MSI were to occur during a cooldown for a SG Tube Rupture.

D - WRONG; EOP 2528 does not have an RNO step to "block and reset" MSI if it occurred because the RCS cooldown was not controlled. The EOP requires MSI be fully actuated.
PLAUSIBLE; Student may confuse the mitigating actions for an undesired MSI during this scenario with that of an undesired MSI during a SG Tube Rupture.

References

MSI Actuation; ARP 2590A-145, R1C0
MSI Block Permitted; ARP 2590A-097, R0C1; EOP 2528, R21C0, St. 23, MSI Block

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 013 013 (SF2 ESFAS) Engineered Safety Features Actuation

Number A2.03 **RO** 4.4 **SRO** 4.7 **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Rapid depressurization

Question #: 38	Question ID: 2018014	Rev. 0	☐ Student Handout?	☐ Lower Order?
Last Edited: 6/14/2018 9:10:24 AM	☑ RO	☐ SRO	Origin: Mod	☐ Past NRC Exam?

Question #: **N/A** **Question ID: 1100019** Rev. 2 ☐ Student Handout? ☐ Lower Order?
☑ **RO** ☐ **SRO** **Origin: Parent** ☑ Past NRC Exam?

PARENT

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 5 psig and slowly rising.
- EOP 2525, Standard Post Trip Actions has been completed.
- 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** Block and reset the MSI actuation, reopen both of the MSIVs, then, due to the loss of control power to PIC-4216, 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, Audit-2018 #43

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 blocked.
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.

Question References not yet listed.

NO Comments or Question Modification History at this time.

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 #: **39** **Question ID: 2018015** **Rev. 0** ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/14/2018 9:10:32 AM ☑ **RO** ☐ **SRO** **Origin: New** ☐ Past NRC Exam?

The plant is at 100% power when a small feedwater leak occurs inside containment on the supply to #1 SG.
The crew has just begun a plant shutdown using OP 2204, "Load Changes".
The containment sump is being pumped to verify the source of the leak.

Which of the following conditions would require entry into an EOP?

☐ **A** Containment temperature has risen to 120°F and a CAR fan trips on overload.

☐ **B** One hour calorimetric power exceeds 2700 MWTH due to fluctuations in feed flow.

☑ **C** Containment sump pump automatically isolates on high containment pressure.

☐ **D** #1 SG feed control must be place in manual control to stabilize feedwater flow.

Question Misc. Info: MP2*ILT 2260, RPS

Justification

A - WRONG; This would require entry into the CTMT Tech. Spec. Due to high temperature and loss of a CAR fan, which would require a plant shutdown, but not a trip and entry into an EOP.
PLAUSIBLE; Student may remember that 120°F is an administrative limit, but not the required actions.

B - WRONG; Exceeding the one hour calorimetric is a reason to take immediate action to stop an inadvertent power rise. However, in this case the calorimetric is being driven high due to fluctuations in feedwater flow, which can occur due to the downpower or feed flow leak. This could be a reason for entering an AOP (Steam Leak), but not a plant trip.
PLAUSIBLE; Student may recall the admin requirement to take immediate action if an uncontrolled plant condition is driving power above the license limit.

C - CORRECT; This implies CTMT cooling capacity has been exceeded by the feedwater leak to the point where CTMT pressure has risen high enough to cause the sump pump to isolate on a CIAS signal. The RPS trip setpoint for High CTMT Pressure is at the same value as the ESAS trigger for CIAS on high CTMT pressure. Therefore, the plant must have also tripped (or must be tripped) at this CTMT pressure and EOP 2525, Standard Post Trip Actions, must then be entered.

D - WRONG; This is an AOP/ARP entry requirement and may lead a crew to trip the plant for conservative reasons, but is not a required plant trip condition.
PLAUSIBLE; Student may consider the loss of feedwater control as a plant trip requirement, which it is, IF manual control didn't stabilize feed flow (which it doesn't imply).

References

ARP 2590A-137, R0C1 and 138, R0C0; AOP 2575, R11C0

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 022 022 (SF5 CCS) Containment Cooling

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)

G2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions."

Question #: **40** | ***Question ID:*** ***55989*** | **Rev.** **1** | ☐ **Student Handout?** | ☐ **Lower Order?**

Last Edited: 6/6/2018 3:17:58 PM | ☑ ***RO*** | ☐ ***SRO*** | ***Origin:*** ***Mod*** | ☐ **Past NRC Exam?**

The plant is in normal operation at 100% power with bus 24E tied to bus 24C,
Then the plant trips due to a Large-Break LOCA.
The RSST was lost at the time of trip.
The following plant conditions exist:

- Bus 24C is de-energized due to a failure of the "A" Emergency Diesel Generator.
- "B" LPSI pump has seized and tripped on overload.
- The crew plans to restore power to the "A" LPSI pump using EOP 2541, Appendix 23, "Restoring Electrical Power".

Which of the following describes some of the specific actions and sequence utilized by Appendix 23 to restore power to the LPSI pump?

☑ **A**
1. Ensure Bus 24E tie to Bus 24C and bus 24D are both OPENED.
2. Energize Bus 24E from Unit 3.
3. Override and reset ESAS UV on Facility 1.
4. Energize bus 24C from bus 24E.

☐ **B**
1. Ensure Bus 24E tied to Bus 24C CLOSED.
2. Ensure 24E tie to bus 24D is OPENED.
3. Override and reset ESAS UV on Facility 1.
4. Energize Bus 24E from Unit 3.

☐ **C**
1. Ensure Bus 24E tie to Bus 24C and bus 24D are both OPENED
2. Energize Bus 24E from Unit 3.
3. Energize bus 24C from bus 24E.
4. Override and reset ESAS UV on Facility 1.

☐ **D**
1. Override and reset ESAS UV on Facility 1.
2. Ensure Bus 24E tied to Bus 24C CLOSED.
3. Ensure 24E tie to bus 24D is OPENED.
4. Energize Bus 24E from Unit 3.

Question Misc. Info: MP2*LOIT ECC-01-C, 2306-9, ECCS

Justification

A - CORRECT; To ensure bus 24E is energized from Unit 3 with absolutely no loads on the bus, Appendix 23 first ensures 24E is isolated from both 24C and 24D before closing the 24E to Unit 3 tie breaker. The ESAS UV signal must be reset

B - WRONG; Bus 24E is required to be isolated from BOTH 24C and 24D to ensure there are no loads on bus 24E when first powering it from Unit 3.
PLAUSIBLE; Student may recognize this as the normal alignment of 24C and 24E, even after a plant trip with a Loss of Normal Power (LNP) from the RSST.

C - WRONG; ESAS UV must be reset before attempting to close the bus 24E to bus 24C tie breaker, due to the UV "load shed" signal present when 24C is de-energized. The signal is designed to isolate 24C (and 24E if tied to it) from all other sources of power or load. With 24E tied to Unit 3, the UV signal would prevent closing the 24E to 24C tie breaker to eliminate the possibility of an EDG attempting to power Unit 3 through the bus ties.
PLAUSIBLE; Student may remember that the UV load shed must be reset before any component can be re-energized because the load shed is designed to remove all "loads" from bus 24C or 24D.

D - WRONG; The ESAS UV signal does not have to be reset before closing the 24E tie to Unit 3 because the "Load Shed" function of ESAS triggered by an LNP does not affect the 24E tie to Unit 3 as long as 24E is not tied to either 24C or 24D.
PLAUSIBLE; Student may remember the UV load shed signal must be reset before power can be restored to 24C, but not the reason and, therefore, when it must be reset.

References

EOP 2541, App 23-N and App 23-D, R3C0

Question #: **40** **Question ID:** **55989** **Rev.** **1** ☐ **Student Handout?** ☐ **Lower Order?**

Last Edited: 6/6/2018 3:17:58 PM ☑ **RO** ☐ **SRO** **Origin:** **Mod** ☐ **Past NRC Exam?**

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 006 006 (SF2; SF3 ECCS) Emergency Core Cooling

Number K2.01 **RO** 3.6 **SRO** 3.9 **CFR Link** (CFR: 41.7)

K2.01 Knowledge of bus power supplies to the following: ECCS pumps

Question #: **N/A** **Question ID: 55989** **Rev. 0** ☐ Student Handout? ☐ Lower Order?
☑ **RO** ☐ **SRO** **Origin: Parent** ☐ Past NRC Exam?

PARENT

The plant is in normal operation at 100% power. The "C" HPSI pump has been declared INOPERABLE. Which of the following bus alignments is necessary to restore a Facility 2 HPSI pump to operable status?

- ☐ **A** Align Bus 24C to Bus 24E. |
- ☐ **B** Align Bus 24A to Bus 24D.
- ☐ **C** Align Bus 24C to Bus 24D.
- ☑ **D** Align Bus 24E to Bus 24D. |

Question Misc. Info: MP2*LOIT*3203 [006 ECC-01-C 3510] (8/16/96) 2306-9, ECCS, NRC, APP

Justification

Align Bus 24E to Bus 24D.

Question References not yet listed.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System**

Number	RO	SRO	CFR Link

Question #: **41** **Question ID: 5000028** **Rev. 1** ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/15/2018 1:42:06 PM ☑ **RO** ☐ **SRO** **Origin: Bank** ☐ Past NRC Exam?

Based on the design functions of the Containment Spray System, what containment condition would be specifically impacted if both CTMT Spray Pumps were out of service during a major accident in containment?

☐ **A** Containment atmosphere heat removal during an Excess Steam Demand event and, Containment wide range sump pH control during a Large-Break LOCA.

☑ **B** Containment atmosphere heat removal during an Excess Steam Demand event and, Containment atmosphere iodine removal during a Large-Break LOCA.

☐ **C** Containment atmosphere hydrogen removal during a Large-Break LOCA and, Containment atmosphere iodine removal during a Large-Break LOCA.

☐ **D** Containment atmosphere hydrogen removal during a Large-Break LOCA and, Containment wide range sump pH control during a Large-Break LOCA.

Question Misc. Info: MP2*ILT CSS, CCS, Tech. Spec. Bases, NRC-2005

Justification

A - WRONG; Sump pH control is provided by trisodium phosphate (TSP) dodecahydrate stored in dissolving baskets located in the containment basement. It functions to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP will get into solution during a LOCA even if containment spray is unavailable and raise the pH of the RCS water blowing out the break and splashing on containment components. As such, Sump pH control is not a function of containment spray.
PLAUSIBLE; Student may know that control of pH is provided by TSP, but believe CTMT Spray is needed to ensure vulnerable CTMT components are "washed" down with pH controlled water.

B - CORRECT; Per TS Basis: The Containment Spray System is more effective than the Containment Cooling System CAR Fans in reducing the temperature of superheated steam inside containment during an ESD event. In addition, the containment spray system provides a mechanism for removing iodine from the containment atmosphere. EOP 2532, Loss Of Coolant Accident, requires CS be operated for a minimum of four hours following a LOCA, even if CTMT pressure reduction would allow it to be secured, to ensure adequate iodine removal.

C - WRONG; Per lesson material: The introduction of highly acidic borated water in a fine mist to the containment will result in the liberation of hydrogen gas in containment. This is produced as a result of the metal-water reaction with aluminum and zinc components. The generation of hydrogen by this mechanism is minimized by controlling the inventory of susceptible metals and by neutralizing the acidity of the water with Trisodium Phosphate.
PLAUSIBLE; Student may remember that the chance of a hydrogen explosion in CTMT is suppose to be reduced by the addition of water vapor to the CTMT atmosphere, and think that this is a design function the plant takes credit for.

D - WRONG: CTMT Spray is not credited for mitigating the hydrogen released to the CTMT atmosphere or the neutralizing of the RCS water blowing on components from the break.
PLAUSIBLE; Student may remember the addition of water vapor to the CTMT atmosphere should reduce the chance of a hydrogen explosion and that the neutralizing of the RCS pH will help prevent damage to wetted components.

References

LP CSS-00-C, R6C0, P3-4; LP PPT-48; TS Bases 3.6.2.1

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 026 026 (SF5 CSS) Containment Spray

Number K3.01 **RO** 3.9 **SRO** 4.1 **CFR Link** (CFR: 41.7 / 45.6)

K3.01 Knowledge of the effect that a loss or malfunction of the CSS will have on the following: CCS

Question #: **42** **Question ID: 2014028** **Rev.** 1 ☐ **Student Handout?** ☑ **Lower Order?**

Last Edited: 6/14/2018 9:11:07 AM ☑ **RO** ☐ **SRO** **Origin: Bank** ☑ **Past NRC Exam?**

Given the following conditions:

- Unit 2 is at 100% when a LOCA occurs.
- EOP 2525 is being implemented.

Which ONE of the following sequences of events would result in reduced Containment Spray Flow (as compared to the same LOCA event without a loss of an MCC)?

☑ **A** Loss of MCC B61, followed 2 minutes later by SIAS and CSAS actuations

☐ **B** Loss of MCC B52, followed 2 minutes later by SIAS and CSAS actuations

☐ **C** SIAS and CSAS actuations, followed 2 minutes later by loss of MCC B51

☐ **D** SIAS and CSAS actuations, followed 2 minutes later by loss of MCC B62

Question Misc. Info: MP2*LOIT NRC-2014

Justification

A - CORRECT; Containment Spray flow is reduced because Containment Spray Isolation valve CS4.1B MOV is powered by B61. This valve fails "AS IS" and the valve is normally closed. The valve will not open on a CSAS.

B - WRONG; Containment Spray system is not affected by a Loss of B52.
Plausible; Student may mix up the MCC powers to CTMT Spray valves (B51/B61 or B52/B62).

C - WRONG; Containment Spray Isolation valve CS 4.1A opens on the CSAS. Its MOV will lose power on the Loss of B51 but the valve fails in the AS IS position. There is no reduction in CS flow since CS 4.1A is already open.
Plausible; Student may be thinking of the loss of power to the CTMT Sump Recirc. Valve, which would substantially affect a SRAS (CTMT Sump Recirc. Signal) when it later occurs.

D - WRONG; Containment Spray system is not affected by a Loss of B62
Plausible; Student may confuse applicable MMC power supplies and CSAS/SRAS impacts.

References

AOP 2503F, R4C0, Att. 3 Load Summary.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 026 026 (SF5 CSS) Containment Spray

Number K2.02 **RO** 2.7* **SRO** 2.9 **CFR Link** (CFR: 41.7)

K2.02 Knowledge of bus power supplies to the following: MOVs

Question #:	43	Question ID:	155052	Rev.	1	☐ Student Handout?	☑ Lower Order?
Last Edited:	6/14/2018 9:11:33 AM	☑ RO	☐ SRO	Origin:	Bank	☐ Past NRC Exam?	

A plant startup is in progress with power stable at 95% for shift turnover. All plant systems and components are operating normally.
Then, the #1 Atmospheric Dump Valve positioner fails causing the ADV to open approximately 25%.

Which one of the following describe immediate actions taken in the control room to stabilize the plant?

☐ **A**
- Start a third condensate pump.
- Lower Main Turbine load to stabilize RCS temperature.
- Verify automatic control of Feed Reg. Valve restoring SG level.

☐ **B**
- Start a third condensate pump.
- Withdraw control rods to stabilize RCS temperature.
- Manually adjust #1 SG feed Reg. valve to stabilize level.

☑ **C**
- Shift both SG feed pumps speed control to manual.
- Lower Main Turbine load to stabilize RCS temperature.
- Manually adjust #1 SG feed Reg. valve to stabilize level.

☐ **D**
- Insert control rods to stabilize RCS temperature.
- Manually adjust SG feed pumps to maintain discharge pressure.
- Verify automatic control of Feed Reg. Valve restoring SG level.

Question Misc. Info: MP2*ILT MSS-01-C, MB-2522, 2316A

Justification

A - WRONG; The steam flow detector is downstream of the ADV, therefore the detector will not "see" the increase in steam flow even though actual steam demand has gone up. The level in #1 S/G will continue to drop.
PLAUSIBLE; Student may feel that starting the third condensate pump will buy time for SG Level Control until turbine load is lowered to stabilize RCS temperature and effectively return steam demand to the pre-event value.

B - WRONG; This will stabilize RCS temperature but it is against administrative guidelines to withdraw control rods to compensate for an abnormal steam demand event, even if less than 100% power.
Plausible; Student may feel the third condensate pump will be needed to compensate for the increase in feedwater demand and recognize that 25% open on an ADV equates to < 2% power.

C - CORRECT; Turbine load is lowered to compensate for the added steam demand seen by the reactor (which is driving the power increase), placing the SGFPs in manual prevents them from responding to any change in steam flow and manual control of feed flow accounts for the additional steam flow from the SG that is not seen by the automatic feedwater control system.

D - WRONG; Power is high due to lowering temp. Lowering nuclear power will only compound the problem. Speed adjust will be of little help as the control system does not see the added steam flow.
Plausible; Student may understand that added steam demand will raise reactor power and remember inserting control rods is usually the best action for an inadvertent addition of positive reactivity. If an inadvertent dilution was the problem, these actions would be correct.

References

AOP 2585, R3Co, St. 6.2 Contingency.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 039 039 (SF4S MSS) Main and Reheat Steam

Number A2.04 **RO** 3.4 **SRO** 3.7 **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

A2.04 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: Malfunctioning steam dump

Question #: **44** **Question ID: 2018016** Rev. 1 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/24/2018 11:09:38 AM ☑ **RO** ☐ **SRO** **Origin: Bank** ☐ Past NRC Exam?

A plant shutdown is in progress and the following conditions exist:

- Power presently at 80%, slowly going down.
- "A" & "B" SG feed pumps in service.
- "A" & "C" condensate pumps in service.
- "B" condensate pump in standby.
- All plant systems and components operating as designed.

Then, C-05 annunciator A-4, "SGFP A TURBINE TRIP", alarms solid.
A Building Service person notifies the Control Room that they bumped the "A" SG feedwater pump (SGFP) causing it to trip.

Which of the following describe the impact and additional required actions per the ARP to mitigate the SGFP trip?

☑ **A** SG levels cannot be maintained above the automatic trip value with one SGFP.
Trip the reactor and turbine and commence EOP 2525, Standard Post Trip Actions.

☐ **B** Maintaining SG levels at this power level may drive SGFP suction pressure too low.
Ensure "B" condensate pump starts, drive rods and reduce turbine load to lower power to </= 50%.

☐ **C** SG levels cannot be maintained above the automatic trip value with one SGFP.
Place main feedwater in manual control and immediately attempt to reset and restart the "A" SGFP.

☐ **D** Maintaining SG levels at this power level may drive SGFP suction pressure too low.
Open the CPF bypass, CNM-2 and verify proper operation of the "B" SG feed pump.

Question Misc. Info: MP2*ILT MFW, 2204, ARP, 2590D-013, MFP trip

Justification

A - CORRECT; Power is > 70%, which per the applicable ARP is not low enough for one SGFP to maintain SG levels. Therefore, the ARP directs the plant be tripped.

B - WRONG; SG levels cannot be maintained with one SGFP if power is > 70%.
PLAUSIBLE; Student may recall that 80% power is the approximate level at which a third condensate pump would be secured, if running, and the problem with loss of a SGFP is the drop in suction pressure. Therefore, starting of a third condensate pump would help.

C - WRONG; It isn't necessary to manually control feed flow, even though the drop in delta-P across the feedwater regulating valves will cause the control system to drive the valves open, compounding the event. Because one SGFP cannot maintain SG levels at this power level, any action taken is only delaying the inevitable. Therefore, a plant trip is required.
PLAUSIBLE; Student may recognize that one SGFP will be challenged to maintain SG levels at this power level, and the control system will drive the control valves open making the situation worse. In the past, the idea was to stop the valves from opening to buy time to restart the tripped pump (which it would do), especially where the pump was not lost due to a failed component or system action.

D - WRONG; ARP 2590D-013 gives actions required for a trip of the "A" SGFP >70% power.
PLAUSIBLE; Student may not remember exact power limit for one SGFP and understand that with the running SGFP providing additional flow that suction pressure will lower and this would necessitate taking immediate operator actions to open the Condensate Demin Bypass valve 2-CNM-2.

References

ARP 2590D-013; R2
(C05; A-5, "SGFP A TURBINE TRIP")

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 059 059 (SF4S MFW) Main Feedwater

Number A2.07 **RO** 3.0* **SRO** 3.3* **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

A2.07 Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Tripping of MFW pump turbine

Question #: **45** **Question ID: 5000034** Rev. 2 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/18/2018 2:04:55 PM ☑ **RO** ☐ **SRO** **Origin: Bank** ☐ Past NRC Exam?

Given the following plant conditions:

- 100% power
- SG levels at setpoint
- Steam flow and feed flow matched

Then, the #2 SG Feed Flow Transmitter, FT-5269A, output fails to 100% of range.

With NO operator actions, which of the following describes the expected plant response?

☐ **A** SG level will lower as the Feed Reg. Valve is driven fully closed by the dominant feed flow signal error.

☑ **B** SG level will lower on the flow mismatch, but the Feed Reg. Valve will not fully close due to level error.

☐ **C** SG level will initially lower but return to setpoint due to level error biasing the dominant steam flow signal.

☐ **D** SG level will not change due to the SG level control system using the lower of the two feed flow signals.

Question Misc. Info: MP2*ILT FWC-01-C RO9 8858, MFW, OP 2385, NRC-2005

Justification

A - WRONG; Output from feed flow transmitters FT-5269A and FT-5269B on the SG2 feed line are averaged for input to the three-element level control. Failing one transmitter high drives the average high. The control system will respond by closing the FRV. The level signal will slowly act on the steam flow signal to moderate the response. However, because of the relatively small error of the feed flow signal and the fact that the signal is STEAM flow dominant, will result in little effect on the level input signal.
PLAUSIBLE; Applicant may think that the feed flow signal error will dominate level control due to the integral action of the system, resulting in SG level continuing to lower.

B - CORRECT; Output from feed flow transmitters FT-5269A and FT-5269B on the SG2 feed line are averaged for input to the three-element level control. Failing one feed flow transmitter high drives the average up a small amount. The control system will respond by closing the FRV in an attempt to maintain Steam Flow and Feed Flow matched. The level signal will slowly act on the steam flow signal to moderate the response. Without operator action, level will decrease a small amount, but soon be turned by both level error and steam flow dominance.

C - WRONG; SG will lower due to the control system response to the mass flow mismatch.
PLAUSIBLE; Applicant may think that based on the control system's integral action, which responds to any deviation from setpoint, the feed flow error will be overcome by the dominant steam flow signal and any SG level error.

D - WRONG; SG will lower due to the control system response to the mass flow mismatch because feed flow signals are averaged, not auctioneered.
PLAUSIBLE; Applicant may mix up which of the inputs to the level control system is auctioneered low (level is) and which are averaged, and based on that believe the failed high feed flow signal will be ignored.

References

OP 2385, R14C0, Discussion

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 059 059 (SF4S MFW) Main Feedwater

Number K1.03 **RO** 3.1 **SRO** 3.3 **CFR Link** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

K1.03 Knowledge of the physical connections and/or cause-effect relationships between the MFW and the following systems: S/GS

Question #: **46** **Question ID:** **53312** **Rev.** 4 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/14/2018 8:21:08 AM ☑ **RO** ☐ **SRO** **Origin:** **Bank** ☐ Past NRC Exam?

The plant was operating at power when a low Steam Generator water level trip occurred due to a loss of BOTH Main Feedwater Pumps.
The following conditions now exist:

- EOP 2525 has been completed.
- An automatic Auxiliary Feedwater Actuation (AFAS) has just occurred.
- Condensate Storage Tank level is 50% (vendor has been notified).
- Both AFW Reg. Valves indicate full open.
- There is NO indicated AFW flow to either SG.

A PEO sent to the AFW system reports the following:

- The "A" and "B" AFW pumps are operating.
- Both pumps are extremely noisy.
- The "A" and "B" AFW pump casings and discharge lines indicate about 265 °F.
- No system valves were found out of position.

Both "A" & "B" AFW pumps were secured.

Which one of the following are actions required to restore AFW flow under the above conditions?

☐ **A** Swap suction to Fire Water, throttle both AFW Reg. Valves to 10% open, then restart both pumps on low flow until cool.

☐ **B** Fully close both of the AFW Reg. Valves, then restart each AFW pump and run on minimum recirculation flow until cool.

☑ **C** Fully close both "A" & "B" AFW pump manual discharge valves, then vent the "A" & "B" pump casings until cool before restarting.

☐ **D** Raise Condensate Storage Tank level to restore pump NPSH, then vent the "A" & "B" pump casings until cool before restarting.

Question Misc. Info: MP2 LOUT, AFW-01-C, MB-2157, **Audit-2018** (Q#46),

Justification

A - WRONG; The pumps must be vented, this action will not sufficiently vent the pumps to purge any entrapped steam.
PLAUSIBLE; Student may consider the Fire Water tanks as a better suction source (higher suction head) due to the low level of the CST.

B - WRONG; The pumps must be vented before attempting to restart if local temperatures are not less than 185°F.
PLAUSIBLE; Student may remember this as the required action, but it is only applicable if local AFW temperatures are less than 185°F.

C - CORRECT - The local indications are that the AFW pumps are steam bound (> 185°F). Per OP-2322, the pumps must be shutdown and vented. Continued operation, in any form, prior to venting has the potential to cause severe damage to the pumps.

D - WRONG; The pump discharge valves must be closed before venting or this action will not sufficiently vent the pumps to purge any entrapped steam.
PLAUSIBLE; Student may believe the cause of the high temperatures is abnormally low level in the Condensate Storage Tank, and not consider backflow leakage through the discharge check valves as the true cause.

References

OP 2322, Auxiliary Feedwater System, Sec. 4.17, "Steam Binding of AFW Pumps".

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 061 061 (SF4S AFW) Auxiliary/Emergency Feedwater

Number K5.05 **RO** 2.7 **SRO** 3.2 **CFR Link** (CFR: 41.5 / 45.7)

K5.05 Knowledge of the operational implications of the following concepts as the apply to the AFW: Feed line voiding and water hammer

Question #: **47** | **Question ID: 2018018** | **Rev. 0** | ☐ Student Handout? | ☐ Lower Order?

Last Edited: 6/14/2018 8:58:58 AM | ☑ **RO** | ☐ **SRO** | **Origin: New** | ☐ Past NRC Exam?

The plant has shut down mid-cycle due to problems with the main turbine.
While shutdown, the "A" charging pump is being tagged out for repairs.
The following conditions exist:

- The "C" charging pump is running with normal letdown flow.
- All power and controls for the "A" charging pump are presently being tagged out.
- The "B" charging pump power supply is being transferred to Facility 1.
- The "A" & "B" charging pump handswitches on C-02/3 are in the "Pull-To-Lock" position.
- The remaining tags for the "A" charging pump are being hung in the plant.
- NO other evolutions are being performed in the plant.

Which of the following control room indications would be <u>unexpected</u> for the above evolutions and indicate that possibly the wrong charging pump has been tagged out?

☑ **A** C-01 annunciator D-40, "FIRE PANELS C09 & C10 TROUBLE" has alarmed.

☐ **B** All three lights on C-02/3 for the "B" charging pump are de-energized.

☐ **C** C-01 annunciator D-41, "SAFE SHUTDOWN PANEL DOORS OPEN" has alarmed.

☐ **D** C-01X blue light is out and the white light is lit for the "A" charging pump.

Question Misc. Info: MP2*LOIT CVC-01-C, 2304E, C-21, CVCS, PLPCS, ESF

Justification

A - CORRECT; This alarm indicates remote panel C-10 is being opened. The "A" charging pump does not have a control switch on C-10, only the "B" & "C" charging pumps do (Facility Two). Therefore, if no other evolutions have been authorized there is no reason for this alarm to come in when tagging out the controls for the "A" charging pump.

B - WRONG; Swapping "B" Charging pump from Facility 2 to Facility 1 involves swapping its power supply, which is a "break-before-make" evolution. Loss of 480 VAC power to the charging pump would cause all C-02 indicating lights for that pump to go out. This is because the 480 VAC powered components get their control power from the 480 VAC supply, downstream of the MCC breaker.
PLAUSIBLE; The student may remember that tagging out the power to the charging pump would deenergize its lights on C-02/3 and believe this is an indication

C - WRONG; This alarm indicates the doors on panel C-21 have been opened. Because the "A" charging pump has a control switch on C-21, this would be an expected alarm for the evolution.
PLAUSIBLE; Student may forget or confuse which remote panels have controls for the specific charging pumps.

D - WRONG; The blue light being out and the white light being energized on C-01X indicate that a component cannot fulfill its design function due to a loss of power (control or otherwise). Racking out and tagging the breaker for the "A" charging pump would cause this very condition. Therefore, this is an expected indication for the given evolution.
PLAUSIBLE; Because this is a "normal" indication for the "B" charging pump, which usually has its handswitch in the "Pull-To-Lock" position, the Student may remember this as being a "normal" light indication for a charging pump. The conclusion being the "A" pump was NOT tagged out.

References

ARP 2590A-160, D-40, "FIRE PANELS C09 & C10 TROUBLE".
AOP 2579A, R12C0, St. 8

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 062 062 (SF6 ED AC) AC Electrical Distribution

Number A4.02 **RO** 2.5 **SRO** 2.8 **CFR Link** (CFR: 41.7 / 45.5 / to 45.8)

A4.02 Ability to manually operate and/or monitor in the control room: Remote racking in and out of breakers

Question #: **48** **Question ID:** **68511** **Rev.** 3 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/17/2018 11:23:41 AM ☑ **RO** ☐ **SRO** **Origin:** **Bank** ☐ Past NRC Exam?

A surveillance run on the 'B' Diesel Generator (DG) is being performed. When removing the DG from service the output breaker trips on reverse power.
While the DG output breaker trip is being evaluated, an LNP occurs on bus 24D.

The following annunciators on C08 are acknowledged and reported to the US:
- DIESEL GEN 13U BKR TRIP
- DIESEL GEN 13U TROUBLE
- DIESEL GEN 13U BKR CLOSING CKT BLOCKED

Which of the following actions must be taken before the "B" DG can power bus 24D?

☐ **A** Have a PEO reset the 'B' DG breaker lockout in the 4160 electrical room.

☐ **B** Have a PEO press the alarm reset button on the 'B' DG skid mounted panel.

☑ **C** Reset the 'B' DG output breaker using the breaker handswitch on C08.

☐ **D** Manually stop and then restart the 'B' DG from the controls on C08.

Question Misc. Info: MP2*LORT*4958 [064 EDG-01-C 3621] (12/5/97) 2346A, EDG

Justification

A - INCORRECT: Have a PEO reset the 'B' DG breaker lockout in the 4160 electrical room. The breaker is reset using the breaker handswitch on C08 and not locally at the breaker.
PLAUSIBLE: Some breaker trips are reset by the relay targets. The DIESEL GEN 13U DIFFERENTIAL LOCKOUT ARP requires resetting relay targets locally at the breaker. It is normal practice to send a PEO to check breaker relay targets.

B - INCORRECT: Have a PEO press the alarm reset button on the 'B' DG skid mounted panel. This will not reset the DG breaker handswitch to allow breaker closure.
PLAUSIBLE: If a LNP condition exists and the diesel engine has been shutdown by an emergency trip, the engine can be restarted by pressing master RESET button on the diesel engine skid mounted gage board. In this case there is a LNP but the diesel engine has not been shutdown. The diesel is running with the output breaker open.

C - CORRECT: Reset the 'B' DG output breaker using the breaker handswitch on C08. As specified in the ARP, the breaker is reset using the output breaker handswitch on C08. A caution in the DG procedure states, "It is acceptable to reset the reverse power relay and the breaker as specified in the ARP and make a second attempt." The ARP specifies IF breaker tripped on reverse power while removing DG from service <u>OR</u> maintenance has been performed on the breaker, PERFORM the following to reset alarm (C-08): 2.1 ENSURE SYN SW 15G-13U-2 (A401) in OFF, 2.2 TURN DG B FDR BKR 15G-13U-2 (A401) to CLOSE. and 2.3 TURN DG B FDR BKR 15G-13U-2 (A401) to TRIP. This resets the breaker. The breaker will now close in response to the LNP and power bus 24D

D - INCORRECT: Manually stop and then restart the 'B' DG from the controls on C08 will not reset the breaker.
PLAUSIBLE: It is reasonable for the examinee to think that if they shut the DG down fully that it will clear all interlocks and reset the breaker.

References

ARP 2590F-144 DIESEL GEN 13U BKR CLOSING CKT BLOCKED
OP 2346C B Emergency Diesel Generator
SP 2613L Diesel Generator Slow Start Operability Test, Facility 2
ARP 2590F-121 DIESEL GEN 13U BKR TRIP
ARP 2590F-142 DIESEL GEN 13U TROUBLE

Facility 2

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 062 062 (SF6 ED AC) AC Electrical Distribution

Question #: **48** **Question ID:** **68511** **Rev.** 3 ☐ **Student Handout?** ☑ **Lower Order?**

Last Edited: 6/17/2018 11:23:41 AM ☑ **RO** ☐ **SRO** **Origin:** **Bank** ☐ **Past NRC Exam?**

Number K4.03 **RO** 2.8* **SRO** 3.1 **CFR Link** (CFR: 41.7)

K4.03 Knowledge of ac distribution system design feature(s) and/or interlock(s) which provide for the following: Interlocks between automatic bus transfer and breakers

Question #: **49** **Question ID: 8073622** Rev. 1 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/19/2018 1:55:32 PM ☑ RO ☐ SRO **Origin: Bank** ☐ Past NRC Exam?

The plant has just tripped from 100% power and the following conditions now exist:
- The BOP noted ALL breaker indicating lights for Bus 24C deenergized a moment before the plant tripped.
- Breaker indicating lights for TCBs #1 and #5 are deenergized.
- The appropriate breaker indicating lights for all other buses are lit.

All other plant equipment is functioning as designed, based on the given plant conditions.

Which of the following describes other system or component responses to the loss of electrical power, WITHOUT operator actions?

☐ **A** At least one Condensate pump and two RCPs were lost.

☐ **B** "A" EDG is running with ONLY the emergency trips available.

☐ **C** The 15G-8T-2 and 15G-9T-2 will trip open on "reverse power".

☑ **D** The "A" and "C" RCPs are running without cooling water.

Question Misc. Info: MP2*LORT*2796 [063 LVD-01-C 972]copy 2345, NRC, APP, NRC-2008

Justification

A - INCORRECT; The Condensate pumps and RCPs are powered from 6.9 kVAC Buses 25A & B, which still have control power and would have transferred to the RSST.
PLAUSIBLE: The student may conclude the given conditions indicate a loss of DC bus 201A (vs. loss of **just** DV-10) .

B - INCORRECT; The "A" EDG will start on a loss of DC (DV-10) with only "overspeed" protection due to the loss of DC control power. Even the emergency trips require Vital DC power.
PLAUSIBLE: The student may remember only "emergency" trips are available when the EDG is automatically started for a loss of power to a vital 4.16 kVAC bus and believe, like many other stand-alone generators, the EDG can self power its own emergency trips.

C - INCORRECT; The turbine trip circuitry is powered by D-11, not DV-10 which powers 24C breakers. Therefore, the main generator output breakers, 8T and 9T, will still receive a trip signal on the plant/turbine trip.
PLAUSIBLE: The student may recall that a loss of DC power will result in the 8T and 9T not receiving a trip signal from the main generator control circuit and, therefore, will not trip until the main turbine starts to slow down and triggers a "reverse power" condition.

D - CORRECT; On a plant trip AC supply power, to 4.16 KVAC busses 24A - 24D and 6.9 KVAC busses 25A & 25B, is designed to automatically transfer from the NSST to the RSST. However, Vital DC load center DV-10 supplies control power to Vital 4.16 KVAC bus 24C. Control power to all of the other KVAC busses comes from other DC sources. Therefore, **only** bus 24C breakers are "frozen" as-is when the plant trips on the closure of both MSIVs (due to the loss of DC power to their "fail-safe" control circuits). This will cause a loss of 24A & 24C on the trip, due to no DC to close the RSST-24C breaker or the "A" D/G output breaker. With no Facility 1 Vital AC power there is no Facility 1 RBCCW cooling to RCPs "A" & "C", which are powered by bus 25A and still running.

References

AOP 2506A R3 Loss of Vital 125 VDC Instrument Panel DV10
125VDC ppt. Lesson Plan

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 063 063 (SF6 ED DC) DC Electrical Distribution

Question #: **49** **Question ID: 8073622** **Rev. 1** ☐ **Student Handout?** ☐ **Lower Order?**

Last Edited: 6/19/2018 1:55:32 PM ☑ **RO** ☐ **SRO** **Origin: Bank** ☐ **Past NRC Exam?**

Number K1.02 **RO** 2.7 **SRO** 3.2 **CFR Link** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

K1.02 Knowledge of the physical connections and/or cause-effect relationships between the DC electrical system and the following systems: AC electrical system

Question #: **50** **Question ID: 8056971** **Rev. 2** ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/7/2018 8:14:43 AM ☑ **RO** ☐ **SRO** **Origin: Mod** ☐ Past NRC Exam?

The "B" Emergency Diesel Generator (EDG) is supplying power to 24D following a plant trip on a loss of offsite power.

Then, the "JACKET COOLING TEMP HIGH" alarm is received on the local panel, C-39.
Service Water (SW) flow meter to the 'B' EDG is reading abnormally LOW.

Which of the following describes what would cause this condition and the result?

☐ **A** An EDG heat exchanger may have become plugged.
The EDG has automatically tripped and the SW flow blockage must be resolved before restart.

☑ **B** An EDG heat exchanger may have become plugged.
Verify EDG heat exchanger D/P is acceptable and reduce loads as needed to clear the alarm.

☐ **C** The EDG SW Bypass Valve may be open.
The EDG has automatically tripped and the Bypass Valve must be closed before restart.

☐ **D** The EDG SW Bypass Valve may be open.
Verify EDG SW flow is restored and the alarm clears after closing the SW Bypass Valve.

Question Misc. Info: MP2*LOIT EDG-01-C, 2346A, EDG

Justification

A - INCORRECT; The EDG automatic trip on High Jacket Cooling Temperature is automatically bypassed when the EDG receives an emergency start signal. The only way the EDG will automatically trip with an emergency start present is if it gets hot enough to lower oil pressure to the trip setpoints or the engine seizes on high temperature.
PLAUSIBLE; Student may remember that this is an EDG automatic trip, but not know it is bypassed on an emergency start.

B - CORRECT; The SW supply to the DG flows through the Air, Lube Oil and Jacket Coolers in series. Therefore, a blockage in any one of them would result in the stated conditions. However, the trip for this alarm is bypassed with an emergency start signal. ARP directed actions are to verify EDG HX DP is acceptable and reduce EDG loads as necessary to clear the alarm.

C - INCORRECT; Although the SW Flow Bypass Valve does not get a close signal when the EDG is automatically started, this valve being open would result in a higher SW flow. This is because the SW flow instrument is upstream of the bypass valve. Therefore, this is NOT indicative of the low SW flow seen on the meter.
PLAUSIBLE; Student may believe this EDG automatic trip is still available at this time and that the SW Flow Bypass Valve also bypasses the SW flow detector.

D - INCORRECT; The EDG SW Bypass Valve being open could result in a high temperature alarm. However, this would NOT result in a lower than normal SW flow indication because the flow detector is upstream of the Bypass Valve.
PLAUSIBLE; Student may believe that the SW Bypass Valve directs SW flow around the flow detector as well as the EDG heat exchangers, and because it is not automatically closed when the EDG starts it could cause the stated conditions.

References

Annunciator C-39, D-2; ARP 2591B-009, R2C3

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 064 064 (SF6 EDG) Emergency Diesel Generator

Number K1.02 **RO** 3.1 **SRO** 3.6* **CFR Link** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

K1.02 Knowledge of the physical connections and/or cause-effect relationships between the ED/G system and the following systems: D/G cooling water system.

Question #: **N/A** **Question ID:** **56971** Rev. 1 ☐ Student Handout? ☐ Lower Order?

☑ **RO** ☐ **SRO** **Origin:** **Parent** ☐ Past NRC Exam?

PARENT

Following a LNP automatic start of the B DG a "JACKET COOLING TEMP HIGH" alarm is received. A PEO reports that the SW flow meter to the B DG is pegged high. Which of the following would cause this condition?

☐ **A** B DG SW strainer is plugged causing the coolant relief valve to open.

☑ **B** B DG SW bypass valve is open and requires operator action to close.

☐ **C** B DG SW supply line has a significantly large pipe leak or rupture.

☐ **D** B DG SW supply is inadvertently cross tied with the A DG SW supply.

Question Misc. Info: MP2*NLO*1971 [064 EDG-04-C 11283] (11/26/97) 2346A, EDG, NRC-2008

Justification

A - Wrong; The SW strainer does NOT have a relief valve to prevent pressure from backing up as it clogs. Therefore, this would cause SW flow to lower.

B - Correct; The SW supply valve will automatically open when the DG is started, but the bypass valve does NOT auto open or close.

C - Wrong; A rupture in the supply line, based on the location of the SW flow instrument, would cause a drop in SW flow.

D - Wrong; Cross-tying the two SW supply headers puts the two supplies in parallel to the single DG supply line. This would NOT raise the supply pressure an appreciable amount, necessary to raise DG SW flow.

Question References not yet listed.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 064 064 (SF6 EDG) Emergency Diesel Generator

Number K1.02 **RO** 3.1 **SRO** 3.6* **CFR Link** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

K1.02 Knowledge of the physical connections and/or cause-effect relationships between the ED/G system and the following systems: D/G cooling water system.

Question #: **51**	**Question ID:** **80593**	**Rev.** **2**	☐ Student Handout?	☑ Lower Order?	
Last Edited: 6/14/2018 8:22:33 AM	☑ **RO**	☐ **SRO**	**Origin:** **Bank**	☐ Past NRC Exam?	

The plant is at 100% power with all plant equipment operating and aligned normally.
Then, bus 22F de-energizes due to a fault in its 4160-to-480 VAC supply transformer.

Based on the guidance of AOP 2503F, Loss of Vital 480 Volt Bus 22F, why does an extended loss of bus 22F potentially require the crew to make the "B" Emergency Diesel Generator (EDG) inoperable?

☐ **A** To ensure the "B" Diesel Generator does NOT start with the room's HVAC unavailable.

☐ **B** To ensure the "B" Diesel Generator does NOT try to start without a fuel oil supply pump.

☐ **C** To prevent the "B" Diesel Generator from trying to start on potentially low starting air pressure.

☑ **D** To prevent the clean fuel oil tank from overflowing onto the "B" Diesel Generator Room floor.

Question Misc. Info: MP2*LOIT 2503F, Bus 22F

Justification

A - WRONG; The EDG rooms have two ventilation systems, one powered from vital power (22F/B61) and one powered form non-vital power. The loss of room ventilation is only a concern if room temperature cannot be maintained below the safety required value of 120℉.
PLAUSIBLE; Student may recall the vital powered EDG room ventilation is considered safety related and auto starts with the EDG to maintain temperature. Therefore, loss of power to the safety related ventilation system would require the EDG be made inoperable.

B - WRONG; Although the EDG's fuel oil supply pump is now de-energized, the fuel oil supply pump is not required as long as the day tank level is > 93%.
PLAUSIBLE; Student may consider that the vast majority of internal combustion engines require a fuel pump to operate, and the loss of this pump would render the engine inoperable.

C - WRONG; The DC compressors are still available and have a pressure setting and capacity high enough to provide adequate starting air pressure. Even with the loss of the Vital DC bus battery charger (22F), the DC bus will still be able to power the EDG DC air compressors long enough to prevent this from being a primary concern.
PLAUSIBLE; The student may not understand the capability of the DC air compressors or consider them unavailable due to DC bus power conservation actions taken on the loss of the battery charger.

D - CORRECT; Fuel oil must be isolated to the "B" EDG to prevent the clean fuel oil tank from overflowing onto the DG Room floor (potential fire and environmental hazard) when the clean fuel oil tank pumps are deenergized.

References

AOP 2503F, R4C0, St. 3.21

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 064 064 (SF6 EDG) Emergency Diesel Generator

Number K6.08 **RO** 3.2 **SRO** 3.3 **CFR Link** (CFR: 41.7 / 45.7)

K6.08 Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Fuel oil storage tanks

Question #: 52	Question ID: 55932	Rev. 1	☐ Student Handout?	☐ Lower Order?
Last Edited: 6/11/2018 1:58:48 PM	☑ RO	☐ SRO	Origin: Bank	☐ Past NRC Exam?

The following radioactive waste conditions exist:

- "CLEAN WASTE DISCHARGE ACTIVITY RIT-9049" annunciator is in alarm (RC-14C)
- Clean Waste RM-9049 "HIGH RAD/FAIL" alarm (C-63 and Local).
- Clean Waste Monitor Tank level is dropping (PPC and local).
- Discharge Final Filter ΔP is reading 15 psid (PPC and local).

Which one of the following conditions, by itself, would allow these indications to occur?

☐ **A** The radwaste discharge throttling valve controller is manually throttling flow.

☐ **B** The radiation monitor has been locally overridden with the "disable key".

☐ **C** Flushing of the radiation monitor is still in progress at the radwaste control panel.

☑ **D** The radwaste discharge valves have been overridden open at the radwaste panel.

Question Misc. Info: MP2 LOIT, ARW-04-C, MB-0633

Justification

A; WRONG; Throttling the discharge valve by manually adjusting the valve controller does not require the valves be overridden open, nor does it stop the rad monitor from closing the valve.
PLAUSIBLE; Examinee may believe the valve must be manually overridden to throttle (like the Aux. Feedwater Reg. Valves) because the normal action of the valve controller is to fully open the discharge valve (like Aux. Feed).

B - WRONG; This only silences the horn on the rad. monitor skid. It does nothing to stop the discharge valves from being closed by the rad monitor.
PLAUSIBLE; Examinee may recognize the rad monitor must be overridden, but not how it is done.

C - WRONG; Flushing the rad. monitor is the only operation that would negate the discharge valve override signal and close the valves, regardless of how they are opened.
PLAUSIBLE; Examinee may believe that opening the rad monitor flush valves could cause a rad monitor failure (low) alarm but not recognize that conditions indicate that the discharge valves must be overridden open.

D - CORRECT; The "Hi Rad/Inst. Fail" annunciator is in alarm and conditions indicate that the discharge valves are still open (ΔP on filter and level dropping). Therefore, the discharge valves MUST be open on override. **[Based on MP2 OE, Inadvertent Aerated Rad. Waste Discharge]**

References

ARP 2590H-027, R0C0; SP 2617D, R1C0, St. 4.1.18

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 073 Process Radiation Monitoring

Number K4.01 **RO** 4.0 **SRO** 4.3 **CFR Link** (CFR: 41.7)

Knowledge of PRM system design feature(s) and/or interlock(s) which provide for the following: Release termination when radiation exceeds setpoint.

Question #: **53** **Question ID: 2018020** Rev. 0 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/14/2018 9:41:05 AM ☑ RO ☐ SRO **Origin: New** ☐ Past NRC Exam?

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

Which of the following evolutions cannot be performed in the control room and require control room operators to utilize Plant Equipment Operators?

☑ **A** Changing RBCCW heat exchanger's Service Water flow control mode in preparation for Thermal Backwashing.

☐ **B** Monitoring for the Ultimate Heat Sink while performing Thermal Backwashing with all instrumentation available.

☐ **C** Isolation of the Service Water supply to a Vital AC Switchgear room cooling coil if the coil suddenly ruptures.

☐ **D** Verify Service Water pump total flow meets Tech. Spec. minimum value following restart of a tripped pump.

Question Misc. Info: MP2*LOIT SW, 2326A, 2325D, Backwash

Justification

A - CORRECT; The Thermal Backwash evolution is expected to raise the temperature of the Niantic Bay water (Ultimate Heat Sink water source) just outside of the Intake Structure. As this is the source for the Service Water System, which then supplies the RBCCW heat exchangers, the RBCCW temperature control valves are shifted to "Summer" mode to minimize the impact of the expected rise heat sink temperature.

B - WRONG; SW inlet temperature must be monitored (and logged) during the performance of Thermal Backwashing, but the procedure states that the PPC is the **preferred** point to be monitored (in the control room).
PLAUSIBLE; The student may remember that digital temperature monitors have been installed at the Service Water System inlet to the RBCCW heat exchangers, for the purpose of verifying the Ultimate Heat Sink (UHS) temperature is within the Tech. Spec. Limit. These monitors display the inlet temperature locally (as well as transmitting the temperature signal to the PPC) and are listed as an alternate place to monitor the UHS temperature.

C - WRONG; If the coil in the switchgear room cooling unit begins to leak, the coffer dam surrounding the cooling unit has a water sensor that will automatically isolate the Service Water supply to the coil. Although a PEO will probably be called to investigate whether this has occurred, local valve operation is not required to isolate the leaking coil.
PLAUSIBLE; Student may remember there is a control board alarm that actuates if moisture is sensed in the coffer dam, but not know that the coil automatically isolates on this alarm.

D - WRONG; Although system flow must be verified following system restoration after a pump trip, it is not necessary to use the instrument required by the Tech. Spec. surveillance.
PLAUSIBLE; The student may recall that digital flow instruments in the turbine building must be used for The Tech. Spec. Surveillance to verify Service Water System total flow .

References

OP 2325D, R15C0, Pg. 19, St. 4.1.9

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 076 076 (SF4S SW) Service Water

Generic K/A Selected

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

Number 2.1.8 **RO** 3.4 **SRO** 4.1 **CFR Link** (CFR: 41.10 / 45.5 / 45.12 / 45.13)

2.1.8 Ability to coordinate personnel activities outside the control room.

Question #: **54** ***Question ID:*** ***1000116*** **Rev.** 2 ☐ **Student Handout?** ☑ **Lower Order?**

Last Edited: 6/17/2018 11:24:59 AM ☑ ***RO*** ☐ ***SRO*** ***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 Redundant Air supply to the Main Feed Water Regulating Valves and Backup Air bottles to the Auxiliary Feed Water Regulating Valves will automatically align to supply air to these valves.

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

Justification

A - INCORRECT: 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 is not correct. There is NO automatic swap to Station Air on a low Containment air pressure. This must be done manually. To accomplish the cross tie of Station Air to Containment Air, manual containment isolation valve SA-19 must be opened and normally closed key locked switch SA-23.1 must be opened.
PLAUSIBLE: Examinee may believe that Containment air loads would receive and "automatic" swap to Station Air since a Containment entry takes a lot of time and most Containment Air loads are safety related.

B - INCORRECT: 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 is not correct. 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.
PLAUSIBLE: Examinee may believe only safety related components will be aligned due to the limited capacity of the SA compressor.

C - CORRECT: 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. The Cross Tie from Station Air to Instrument Air, 2-SA-10.1 and the Station Air Header Isolation, 2-SA-11.1 automatically operate off of the Instrument Air Receiver Tank, T51 pressure. When Instrument Air Receiver Tank, T51 pressure falls below 85 psig the Cross Tie from Station Air to Instrument Air, 2-SA-10.1 opens and the Station Air Header Isolation, 2-SA-11.1 closes. This is done so all of the Station Air capacity is supplied to Instrument Air in an attempt to maintain Instrument Air Header pressure.

D - INCORRECT: The Redundant Air supply to the Main Feed Water Regulating Valves and Backup Air bottles to the Auxiliary Feed Water Regulating Valves will automatically align to supply air to these valves. Although there is a Redundant Air supply to the Main Feed Water Regulating Valves, it is only a second piping run supplied from the same Instrument Air Receiver Tank. This Redundant Air supply is always aligned, therefore it does not automatically align. Backup Air bottles to the Auxiliary Feed Water Regulating Valves will not automatically align to supply air to these valves. The Backup Air bottles to the Auxiliary Feed Water Regulating are always aligned. Therefore there is NO automatic alignment of the backup air system to the Auxiliary Feed Water Regulating Valves.
PLAUSIBLE: The Redundant Air supply to the Main Feed Water Regulating Valves is designed to limit the chance of an IA header rupture causing a loss of FRV control. The examinee can reasonably think that a backup air system could automatically align to prevent a plant trip as a result of a loss of IA to the Main Feed Water Regulating Valves. Other important components have air accumulators for this purpose. Also the examinee could think the Backup Air bottles to the Auxiliary Feed Water Regulating Valves would automatically align since the Auxiliary Feed Water Regulating Valves are safety related.

References

AOP 2563, Loss of Instrument Air
ISA-00-C, Lesson Text, Station Air & Instrument Air Systems

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 078 078 (SF8 IAS) Instrument Air

Number A3.01 **RO** 3.1 **SRO** 3.2 **CFR Link** (CFR: 41.7 / 45.5)

A3.01 Ability to monitor automatic operation of the IAS, including: Air pressure

Question #: **55** **Question ID: 2014031** Rev. 2 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/17/2018 11:25:32 AM ☑ RO ☐ SRO **Origin: Mod** ☐ Past NRC Exam?

Select the choice below that describes equipment actuations that are expected to result from depressing BOTH Containment Isolation Actuation Signal pushbuttons on C-01.

☐ **A** Reactor Coolant Letdown Valve, CH-515 closes.

☑ **B** Any running Main Exhaust Fan, F-34A/B/C, stops.

☐ **C** Control Room Filter Fan/Damper, F32A and F32B Start/Open.

☐ **D** Main Steam Isolation Valves, MS-64A and MS-64B close.

Question Misc. Info: MP2*LOIT NRC-2014. Updated for 2018 NRC exam. Replaced the correct answer. Modified two distracters. Completely replaced the third distracter.

Justification

A - INCORRECT: Reactor Coolant Letdown Valve, CH-515 closes is not correct. Manual actuation of CIAS does not activate SIAS and EBFAS. SIAS closes the Reactor Coolant Letdown valve CH-515.
PLAUSIBLE: Because automatic actuation of SIAS causes a EBFAS and CIAS, applicant may believe that manual actuation of CIAS will cause a SIAS and EBFAS. The examinee could also believe that CH-515 is a CIAS valve since the other Letdown Isolation valve CH-516 is a CIAS actuated valve.

B - CORRECT: Main Exhaust Fans, F-34A/B/C stop is correct.

C - INCORRECT: Control Room Filter Fan/Damper F32A and F32B Start/Open is not correct. The Control Room Filter Fan/Damper F32A and F32B Start/Open is actuated by EBFAS.
PLAUSIBLE: It is reasonable for the examinee to think that a CIAS would place CRAC in recirculation since CIAS stops radiation from reaching the public. Applicant may also believe that manual initiation of CIAS also initiates EBFAS because automatic initiation of SIAS also actuates EBFAS.

D - INCORRECT: Main Steam Isolation Valves, MS-64A and MS-64B close is not correct. These close on a MSI signal and not a CIAS signal.
PLAUSIBLE: Main Steam Isolation Valves, MS-64A and MS-64B are containment isolation valves. It is reasonable to believe that containment isolation valves would close on a CIAS. Also MSI and CIAS have the same Containment Pressure setpoint of 4.42 psig. Applicant may believe that CIAS also isolates these valves because they allow steam to exit containment.

References

ESA-01-C, Lesson Text, Engineered Safety Features Actuation System

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 103 103 (SF5 CNT) Containment

Number A3.01 **RO** 3.9 **SRO** 4.2 **CFR Link** (CFR: 41.7 / 45.5)

A3.01 Ability to monitor automatic operation of the containment system, including: Containment isolation

Question #: **N/A** **Question ID: 2014031** Rev. 1 ☐ Student Handout? ☐ Lower Order?

☑ **RO** ☐ **SRO** **Origin: Parent** ☑ Past NRC Exam?

PARENT Select the choice below that describes equipment actuations that are expected to result from depressing BOTH Containment Isolation Actuation Signal pushbuttons on C-01.

☑ **A** STOP: F23 Containment Purge Supply Fan
CLOSE: 2-CH-089 Regen HX Disch to Letdown HX and 2-RB-402 L/D HX RBCCW TCV

☐ **B** CLOSE: 2-CH-515 Reactor Coolant Letdown Valve
START: F25A/B Enclosure Building Filtration Fans
OPEN: 2-EB-52 & 42 Enclosure Bldg Fans 25A & B Disch Dampers
CLOSE: RCP Controlled Bleed Off Isolation Valves, 2-CH-505, 2-CH-506 and 2-CH-198.

☐ **C** CLOSE: 2-CH-515 Letdown Isolation, 2-CH-501 VCT Outlet Valve
START: Boric Acid Pumps P19A and P19B
CLOSE: 2-CH-089 Regen HX Disch to Letdown HX and 2-RB-402 L/D HX RBCCW TCV

☐ **D** CLOSE: 2-MS-64A/B MSIVs 1&2 and 2-MS-65A/B MSIV Bypasses 1&2
CLOSE: 2-FW-42A/B SG 1&2 Feedwater Block Valves

Question Misc. Info: MP2*LOIT NRC-2014

Justification

A - CORRECT: Manual activation of CIAS from the control room causes these equipment actuations.

B - Incorrect: Manual actuation of CIAS does not activate SIAS and EBFAS. SIAS closes CH-551 Reactor Coolant Letdown valve, EBFAS starts F25A/B Enclosure Building Filtration Fans and opens 2-EB-42. CIAS does close RCP Controlled Bleed Off Isolation Valves, 2-CH-505, 2-CH-506 and 2-CH-198.
Plausible: Because automatic actuation of SIAS causes a EBFAS and CIAS, applicant may believe that manual actuation of CIAS will cause a SIAS and EBFAS.

C - Incorrect: SIAS closes CH-515 Reactor Coolant Letdown, CH-501 VCT Outlet Valve and starts Boric Acid Pumps P19A and P19B. CIAS closes 2-CH-089 Regenerative Heat Exchanger Discharge to Letdown HX and 2-RB-402 Letdown Heat Exchanger RBCCW TCV.
Plausible: Applicant may believe that manual initiation of CIAS also initiates SIAS because automatic initiation of SIAS initiates CIAS.

D - Incorrect: All of these components are activated by a MSI.
Plausible: MSI and CIAS have the same Containment Pressure setpoint of 4.42 psig. Applicant may believe that CIAS also isolates these valves because they allow steam to exit containment.

Question References not yet listed.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 103 103 (SF5 CNT) Containment

Number A3.01 **RO** 3.9 **SRO** 4.2 **CFR Link** (CFR: 41.7 / 45.5)

A3.01 Ability to monitor automatic operation of the containment system, including: Containment isolation

Question #: **56** **Question ID: 2018021** Rev. 0 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/17/2018 11:26:19 AM ☑ RO ☐ SRO **Origin: New** ☐ Past NRC Exam?

Which of the following is a condition that does NOT meet the Natural Circulation criteria of EOP2528, Loss of Offsite Power/Loss of Forced Circulation AND the procedural action required to restore the parameter to within the criteria for Natural Circulation?

☐ **A** T-hot and T-cold are lowering, take action to raise feedwater flow.

☐ **B** T-hot and CET ΔT is 5 °F, take action to lower the steaming rate.

☑ **C** CET subcooling is 25 °F, take action to raise RCS pressure with pressurizer heaters.

☐ **D** Steam Generator Level is 40% and lowering, take action to raise feedwater flow.

Question Misc. Info: New question for 2018 NRC exam.

Justification

A - INCORRECT: T-hot and T-cold are lowering, take action to raise feedwater flow is not correct. T-hot and T-cold are lowering MEETs natural circulation criteria.
PLAUSIBLE: The examinee may think that only T-hot should be lowering since with power lowering over time that ΔT will decrease and that T-cold will stay constant. Also they could think that more feedwater is needed since with both Tc and Th lowering that steam flow has increased and that would require greater feed flow.

B - INCORRECT: T-hot and CET ΔT is 5 °F, take action to lower the steaming rate is not correct. T-hot and CET ΔT 5 °F MEETs the TH AND CET temperature ΔT is less than 10 °F criteria. If this ΔT was greater that 10 °F then taking action to lower the steaming rate would be correct.
PLAUSIBLE: The examine may think that CETs and T-hot temperatures should be the same. There is really no reason for them to be different and they are normally with a degree or two on the simulator. And taking action to lower the steaming rate would be the correct action if the T-hot and CET ΔT criteria was not met.

C - CORRECT: CET subcooling is 25 °F, take action to raise RCS pressure with pressurizer heaters. CET subcooling 25 °F does not meet the natural circulation criteria of CET subcooling is greater than OR equal to 30 °F. One of the response not obtained actions is to OPERATE pressurizer heaters and spray to maintain CET subcooling above the minimum operating limit of the RCS P/T curve, which is what is provided in the answer to take action to raise RCS pressure with pressurizer heaters.

The Millstone Unit 2 Loss Of Offsite Power/Loss of Forced Circulation procedure EOP 2528 contains Single Phase Natural Circulation criteria and provides for actions to take (response not obtained) if the criteria is not met.

Criteria for Single Phase Natural Circulation is; CHECK natural circulation flow in at least ONE loop by ALL of the following: Loop ΔT is less 55 °F, TH AND TC are constant OR dropping, CET subcooling is greater than OR equal to 30 °F, TH AND CET temperature ΔT is less than 10 °F.

Response not obtained actions are: ENSURE least affected steam generator pressure is less than saturation pressure for existing Tc, CONTROL least affected steam generator feeding AND steaming to establish natural circulation, and OPERATE pressurizer heaters and spray to maintain CET subcooling above the minimum operating limit of the RCS P/T curve. REFER TO EOP 2541, Appendix 2, “Figures,” Figure 1.

D - INCORRECT: Steam Generator Level is 40% and lowering, take action to raise feedwater flow is not correct. Steam Generator Level is 40% and lowering is NOT a natural circulation criteria. It is part of the criteria for HPSI Throttle stop criteria.
PLAUSIBLE: The examinee could think that maintaining Steam Generator Level was a natural circulation criteria since maintaining a heat sink is required for natural circulation. This is part of the HPSI throttle stop criteria. And taking action to raise steam generator level would be the appropriate response to lowering steam generator level.

References

EOP 2528, Millstone Unit 2 Loss Of Offsite Power/Loss of Forced Circulation R021-00

Question #: 56	**Question ID: 2018021**	**Rev.** 0	☐ Student Handout?	☐ Lower Order?
Last Edited: 6/17/2018 11:26:19 AM	☑ **RO**	☐ **SRO**	**Origin: New**	☐ Past NRC Exam?

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 002 002 (SF2; SF4P RCS) Reactor Coolant

Number A2.03 **RO** 4.1 **SRO** 4.3 **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.5)

A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of forced circulation

Question #: **57** **Question ID:** **8000067** **Rev.** 1 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/14/2018 9:11:49 AM ☑ **RO** ☐ **SRO** **Origin:** **Mod** ☐ Past NRC Exam?

Channel "X" Pressurizer Level and Pressure Control are selected as the controlling channels.
The plant is tripped when bus 24D deenergizes due to a bus fault.

The pressurizer is stabilized at 40% level and 2250 psia, when VR-21 deenergizes on failure of its main breaker.
The RO takes the applicable actions to restore the available pressurizer heaters.

How would the pressurizer heaters automatically respond under these conditions, if pressurizer level were to suddenly rise to 44% due to an RCS insurge?

☑ **A** Backup Heaters: ALL four groups deenergized.
Proportional Heaters: Group 1 only energized at MAXIMUM output.

☐ **B** Backup Heaters: ALL four groups deenergized.
Proportional Heaters: Group 1 only energized at MINIMUM output.

☐ **C** Backup Heaters: Groups 1 and 3 energized.
Proportional Heaters: Group 1 only energized at MAXIMUM output.

☐ **D** Backup Heaters: Groups 1 and 3 energized.
Proportional Heaters: Group 1 only energized at MINIMUM output.

Question Misc. Info: MP2*LOIT 2304A, PLPCS, VR-21, 2504B, NRC-2008, NRC-2016

Justification

A - CORRECT; PZR level rising to 44% on the RCS insurge will cause the level control system to energize all available Backup (B/U) Heaters and drive the Proportional Heaters to maximum output, even though the insurge will drive pressurizer pressure above setpoint. However, the B/U Heaters Groups 1 and 3 will be prevented from energizing, because the High PZR Pressure bistable that normally triggers at 2350 psia will fail to the "activate" mode on loss of VR-21. The bistable is designed to override any signal to energize the B/U heaters and prevent an over pressure condition, but it only affects the B/U heaters.

C - WRONG; The B/U heaters are prevented from energizing by the loss of VR-21.
PLAUSIBLE; Examinee may recall the effect of a PZR level insurge on the heaters, but not know the effect of VR-21 loss.

B - WRONG; The impact of the PZR insurge to 44% on the PZR Control System will override the effect of the rise in pressure to 2310 psia.
PLAUSIBLE; Examinee may focus on the effect of VR-21 loss on the backup heaters and assume the Proportional heaters would then respond accordingly to the expected rise in pressure.

D - WRONG; The B/U heaters are prevented from energizing by the loss of VR-21 and the insurge to 44% will cause the Proportional heaters to go to maximum output.
PLAUSIBLE; Examinee may believe the insurge bistable only affects the non-vital B/U heaters and that it would function because the High Pressure bistable is deenergized. This would leave the vital Proportional heaters to respond to the expected rise in pressure.

References

OP 2204, R39C0, Att. #4;
AOP 2504B, R5C0, Discussion.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 011 011 (SF2 PZR LCS) Pressurizer Level Control

Number K2.02 **RO** 3.1 **SRO** 3.2 **CFR Link** (CFR: 41.7)

K2.02 Knowledge of bus power supplies to the following: PZR heaters

Question #: **N/A** | ***Question ID:* 8000067** | **Rev. 0** | ☐ **Student Handout?** | ☐ **Lower Order?**

☑ ***RO*** | ☐ ***SRO*** | ***Origin:* Parent** | ☑ **Past NRC Exam?**

PARENT

The following initial plant conditions exist:

- 100% steady-state.
- Channel "Y" Pressurizer Level and Pressure Control selected as the controlling channels.

Then, VR-21 deenergizes.

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

☐ **A** Channel "Y" pressurizer pressure input would fail low, causing pressure control to slowly raise actual pressurizer pressure.

☐ **B** Channel "Y" pressurizer level input would fail low, causing pressurizer level control to slowly raise actual pressurizer level.

☐ **C** Pressurizer backup heaters would deenergize if on, RCS pressure would stabilize on the proportional heater output.

☑ **D** All pressurizer heaters would deenergize, spray valve bypass flow would cause RCS pressure to continue to lower.

Question Misc. Info: MP2*LOIT 2304A, PLPCS, VR-21, 2504B, NRC-2008, NRC-2016

Justification

A - WRONG; PZR pressure input is powered by VA-20 (VIAC), NOT VR-21, a non-vital power supply.
Plausible; Examinee may believe pressure and level control circuits are powered by the same type of power.

B - WRONG; Although the Ch. "Y" PZR level control circuit is normally powered from VR-21, the level transmitter (input) is powered by a VIAC (VA-20).
Plausible; Examinee may assume the PZR level transmitting circuit is powered by the same source as the level control circuit.

C - WRONG; The loss of VR-21 would cause a loss of ALL heaters due to the impact on the low level trip circuit.
Plausible; Examinee may focus on the effect of VR-21 on the backup heaters due to the impact on the high pressure trip circuit, which trips all Backup Heaters (only) and prevents them from being energized.

D - Correct; The Pressurizer Heater Selector switch is normally in the "Both" position, which means a loss of VR-11 OR VR-21 will cause all PZR heaters to deenergize due to the failure of the heater low level cutout circuit. The recovery of the heaters requires the operators to de-select the failed/de-energized circuit (select Ch. "X" only) and reclose both Proportional heater breakers.

References

AOP 2504B

NO Comments or Question Modification History at this time.

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

Number K3.01 **RO** 3.8 **SRO** 3.9 **CFR Link** (CFR: 41.7 / 45.6)

Knowledge of the effect that a loss or malfunction of the PZR PCS will have on the following: RCS

Question #: **58** **Question ID:** **78148** Rev. 5 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/14/2018 8:26:51 AM ☑ **RO** ☐ **SRO** **Origin:** **Bank** ☐ Past NRC Exam?

The plant at 100% power, steady state, with all CEAs fully withdrawn. An RO is performing the Tech. Spec. surveillance for CEA motion verification. The RO has inserted Group 7 CEA #1 two steps when it slips to the 174 steps withdrawn position.
The following conditions now exist:

- The RED light on the Core Mimic for CEA #1 remains energized.
- I&C has verified the reed switch for CEA #1 red mimic light is stuck closed.
- All other CEDS controls and indications are functioning as designed.

Which one of the following describes what must now be done before the CEDS will allow CEA #1 to be returned to the fully withdrawn position?

☑ **A** The Upper Electrical Limit must be reset or cleared.

☐ **B** The stuck reed switch input to CEAPDS must be blocked.

☐ **C** The pulse counts must be reset to match reed position.

☐ **D** The CEA Motion Inhibit must be bypassed by the RO.

Question Misc. Info: MP2*LORT CEDS, Interlocks, UEL, Reeds

Justification

A - CORRECT; The Upper Electrical Limit (UEL) is now armed and will not allow any further withdrawal of CEA #1 until "cleared" by I&C by lifting a lead in the CEDS Logic Cabinets.

B - WRONG; The reed switches that trigger the Upper and Lower Electrical Limits are separate from the reed switches that feed into CEAPDS.
PLAUSIBLE; Student may remember that a failed reed switch input into CEAPDS can be blocked to prevent the triggering of a rod motion interlock from a input failure.

C - WRONG; The PPC would indicate the CEA has not moved past where it was driven (178 steps) and has no effect on the UEL. However, its pulse count indication (PPC) must be reset to the actual CEA position (174 steps) by procedure, before the CEA is returned to its fully withdrawn position.
PLAUSIBLE; Student may remember that when fully withdrawing CEAs in group mode, the PPC would stop group motion when the CEAs were at 176 steps out. This is when the Upper Core Stop is triggered by pulse counts on the PPC.

D - WRONG; CMI is not in active for these conditions (CEA did not slip far enough). Also, bypassing the CMI has no effected on the "failed on" UEL, caused by the stuck reed switch.
PLAUSIBLE; Student may know the UEL interlock will prevent CEA withdrawal but not exactly how it functions.

References

OP 2302A, R20C0, Att. 5

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 014 014 (SF1 RPI) Rod Position Indication

Number K1.01 **RO** 3.2* **SRO** 3.6 **CFR Link** (CFR: 41.3 to 41.9 / 45.7 to 45.8)

K1.01 Knowledge of the physical connections and/or cause-effect relationships between the RPIS and the following systems: CRDS

Question #: **59** **Question ID: 2018023** Rev. 0 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/24/2018 11:13:35 AM ☑ RO ☐ SRO **Origin: New** ☐ Past NRC Exam?

The plant is being shutdown for a refueling outage. Which power indication is used and what indications will the operators have that the turbine trip is bypassed on the Reactor Protection System?

☐ **A** Highest of power range nuclear instruments and Delta T power, power range monitor LINEAR Power LED light goes out when the LED Reset is toggled.

☐ **B** Highest of Power range nuclear instruments and Delta T power, power range monitor LINEAR Power LED light comes on solid when the LED Reset is toggled.

☑ **C** Power range nuclear instruments, power range monitor LINEAR Power LED light goes out when the LED Reset is toggled.

☐ **D** Power range nuclear instruments, power range monitor LINEAR Power LED light comes on solid when the LED Reset is toggled.

Question Misc. Info: New question for 2018 NRC ILT exam.

Justification

A - INCORRECT: Highest of power range nuclear instruments and Delta T power, power range monitor LINEAR Power LED light comes on solid when the LED Reset is toggled is not correct. Only nuclear instruments are inputs to this bistable for power indication and when the LED RESET is toggled the LEDs are not lit.
PLAUSIBLE: It is reasonable for the examinee to think that power would be from Q power, the highest of NI and ΔT power, because that is what is used in most RPS trips that have a power input. Whether the LED light is on solid or off is just something they need to know.

B - INCORRECT: Highest of Power range nuclear instruments and Delta T power, power range monitor LINEAR Power LED light comes on solid when the LED Reset is toggled is not correct. Only nuclear instruments are inputs to this bistable for power indication. When the LED RESET is toggled the LEDs are not lit.
PLAUSIBLE: It is reasonable for the examinee to think that power would be from Q power, the highest of NI and ΔT power, because that is what is used in most RPS trips that have a power input.

C - CORRECT: Power range nuclear instruments, power range monitor LINEAR Power LED lit goes out when the LED Reset is toggled is correct. ARP 2590C-055, TURBINE AND LCL PWR DENSITY TRIP BYPASSED, states IF power is less than 15% AND linear power light clears (not lit power range monitors) on at least 3 channels, the following trips are inhibited: Turbine Trip and Local Power Density Trip.

When reactor power is < 15% on nuclear instruments the turbine trip is inhibited. When power is < 15% and linear power lights clear (not lit on the power range monitors) on at least three channels, the turbine trip is inhibited (bypassed). The examinee needs to understand that when power is lowered to <15% (as indicated on the linear range nuclear instruments) that the LINEAR POWER LEDs will flash when the bistable automatically resets. The bistable is reset when flashing. The operator confirms the turbine trip is bypass when the LED reset is toggled and the linear power LED lights go out. Three channels are required for the trip to be bypassed.

D - INCORRECT: Power range nuclear instruments, power range monitor LINEAR Power LED light comes on solid when the LED Reset is toggled is not correct. When the LED RESET is toggled the LEDs are not lit.
PLAUSIBLE: It is reasonable for the examinee to not know if the LED light will be on or off.

References

ARP 2590C-055, TURBINE AND LCL PWR DENSITY TRIP BYPASSED
NIS-01-C Lesson Plan, Nuclear Instrumentation System
OP 2205, Plant Shutdown

NO Comments or Question Modification History at this time.

Question #: **59** **Question ID: 2018023** **Rev. 0** ☐ **Student Handout?** ☑ **Lower Order?**

Last Edited: 6/24/2018 11:13:35 AM ☑ **RO** ☐ **SRO** **Origin: New** ☐ **Past NRC Exam?**

NRC K/A System/E/A **System** 015 015 (SF7 NI) Nuclear Instrumentation

Number A4.03 **RO** 3.8 **SRO** 3.9 **CFR Link** (CFR: 41.7 / 45.5 to 45.8)

A4.03 Ability to manually operate and/or monitor in the control room: Trip bypasses

Question #: **60** | ***Question ID:* 2018022** | Rev. 0 | ☐ Student Handout? | ☑ Lower Order?

Last Edited: 6/14/2018 8:27:02 AM | ☑ ***RO*** | ☐ ***SRO*** | ***Origin:* New** | ☐ Past NRC Exam?

Plant power has been lowered for Main Turbine Control Valve testing.
The following conditions presently exist:

- Power is at 90% and stable.
- RCS Tavg and pressurizer level are both on program.
- "A" & "C" Charging Pumps operating.
- Forcing Pressurizer Sprays.
- All plant systems and components functioning as designed.

Then, RCS Loop 2 Thot control RTD signal from TE-121X, fails to 515℉.
Several annunciators on C-02/3 and C-04 are in alarm, including C-02/3, Pressurizer Level Hi/Lo for both channels.

Which of the following is the procedurally required initial actions to stabilize pressurizer level?

☐ **A** Place the "A" & "C" Charging Pumps in "Pull-To-Lock" and isolate Letdown flow.

☐ **B** Go to the Reactor Reg. Foxboro IA control screen and bypass the TE-121X input.

☑ **C** Shift Letdown Flow Controller, HIC-110, to manual and stabilize pressurizer level.

☐ **D** Return the bias on Letdown Flow Controller, HIC-110, to the normal at-power level.

Question Misc. Info: MP2*LORT*4051 [011 PLC-01-C 4814] (10/3/97) 2304A, PZR, RRS

Justification

A - WRONG: This is the Immediate Operator Action for a loss of one of the four control power sources to the CVCS and Rx Reg. systems.
PLAUSIBLE: Student may recall the dramatic effect this would have on the CVCS and remember AOP 2585, IOA, has guidance for isolating CVCS when the system's operation is challenged by a plant malfunction.

B - WRONG: This would cause the CVCS to have another dramatic response, similar to the one it just had when the RTD failed. This is why the governing ARP has letdown placed in manual control before the failed sensor input is bypassed.
PLAUSIBLE: Student may understand what has to eventually be done to correct the problem and may immediately jump to that action.

C - CORRECT: At 90% power pressurizer level setpoint is still 65%. With this failure, Tavg would be (515℉ + 589℉ + 543℉ + 543℉) / 4 = ~548℉. The pressurizer level setpoint for Tavg at 548℉ is ~51%. Therefore, the failure will reduce the level setpoint well below actual pressurizer level, causing the CVCS control systems to override the manual start of the backup charging pump ("A") and shut it down. It will also cause Letdown flow to go to maximum, effectively draining the pressurizer in an attempt to restore level to "setpoint".

D - WRONG: HIC-110 bias was adjusted to match Charging and Letdown flows when a second charging pump ("A") was manually started. However, adjusting the bias too quickly in an attempt to restore Letdown flow to normal could easily cause a severe pressure transient in the system and lift a relief valve. Therefore, the procedure directs manually controlling the output of HIC-110 to stabilize pressurizer level.
PLAUSIBLE: Student may recognize that the backup charging pump that was manually started has secured on the failed instrument and that the bias was adjusted only because it was started.

References

AOP 2585, R3C0, Immediate Operator Actions, Sec. 9, "Pressurizer Level Control Malfunctions".

NO Comments or Question Modification History at this time.

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 2.4.31 **RO** 4.2 **SRO** 4.1 **CFR Link** (CFR: 41.10 / 45.3)

G2.4.31 Knowledge of annunciator alarms, indications, or response procedures.

Question #: **61** **Question ID: 2018043** **Rev. 0** ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/20/2018 9:11:45 PM ☑ **RO** ☐ **SRO** **Origin: New** ☐ Past NRC Exam?

Which of the following signals will cause an automatic isolation of the Hydrogen Purge System?

- ☑ **A** Either a CIAS signal or high radiation signal from CTMT High Range Rad. Monitors.
- ☐ **B** Either a CIAS signal or a high radiation signal from CTMT Atmosphere Rad. Monitors.
- ☐ **C** Only a CIAS signal.
- ☐ **D** Only a high radiation signal from CTMT High Range Rad. Monitors.

Question Misc. Info: MP2*ILT RM-8240, H2 Purge

Justification

A - CORRECT; The CTMT purge valves will auto close on a CIAS or a HI Rad/Failure of the CTMT High Rad Monitors, RM-8240 or RM-8241.

B - WRONG; CIAS will close the purge valves, but only a signal from the CTMT High Range Rad Monitors will auto close the H2 Purge valves.
Plausible; Student may confuse which rad monitors auto isolate which purge system.

C - WRONG; CIAS will close the valves but it is not the only signal that can do so.
Plausible; Student may recognize the valves must be closed in an accident to ensure CTMT integrity is maintained and, therefore, assume this is the key signal.

D - WRONG; The CTMT Atmosphere Rad Monitors will auto close the CTMT Purge Valves, not the CTMT H2 Purge Valves.
Plausible; Student may confuse which rad monitors auto isolate which purge system.

References

ARPs 2590A-119, 2590A-137, 2590A-138

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 029 029 (SF8 CPS) Containment Purge

Number K4.03 **RO** 3.2* **SRO** 3.5 **CFR Link** (CFR: 41.7)

K4.03 Knowledge of design feature(s) and/or interlock(s) which provide for the following: Automatic purge isolation

Question #: **62** **Question ID:** **55162** Rev. 6 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/11/2018 3:05:10 PM ☑ RO ☐ SRO **Origin:** **Mod** ☐ Past NRC Exam?

The Spent Fuel Pool Cooling (SFPC) system is in a normal lineup, when the following Control Room overhead annunciators are received:

- SFP CLG PUMP SUCTION FLOW LO
- SFP LEVEL LO

The cause of the alarms is determined to be lowering SFP level due to a leak between the SFPC Pumps and the SFPC heat exchangers.

Assuming the SFPC check valves all operate with no leakage, which of the following describes the impact of this leak?

☐ **A** The SFP will drain to the discharge line siphon breaker and the running SFP cooling pump will automatically trip on low suction pressure.

☐ **B** The SFP will drain to the discharge line siphon breaker and the running SFP cooling pump must be manually secured.

☐ **C** The SFP will drain to the suction line siphon breaker and the running SFP cooling pump will automatically trip on low suction pressure.

☑ **D** The SFP will drain to the suction line siphon breaker and the running SFP cooling pump must be manually secured.

Question Misc. Info: MP2*LOIT*2502 [033 SFP-01-C 7324] (2/5/2001) 2305, SFP, NLIT, NRC-2014

Justification

A - WRONG: The question stem states to assume check valves operate with no leakage. Therefore the suction line siphon breaker will stop the SFP level loss. Also, the SFPC pumps do not have an automatic trip on low suction pressure.
PLAUSIBLE; Because the discharge line siphon breaker is designed to mitigate a leak in the discharge of the pumps, the student may not recognize the effect of the check valves on a leak in that specific location, and also assume that a low suction pressure trip would be logical due to the use of siphon breakers to stop pool level loss.

B - WRONG: The question stem states to assume check valves operate with no leakage. Given the leak location between SFP Cooling Pumps discharge and the SFP Cooling Heat Exchangers, the Downstream Check Valves 2-RW-8 and 2-RW-10 would prevent any back-leakage from the pool through the discharge line, without reliance on the discharge line siphon breaker.
PLAUSIBLE; Because the discharge line siphon breaker is designed to mitigate a leak in the discharge of the pumps, the student may not recognize the effect of the check valves on a leak in that specific location.

C - WRONG: Although the leak will be stopped by the suction line siphon breaker, the SFPC pumps will not trip on low suction pressure.
PLAUSIBLE; Student may believe a pump protective circuit of some kind exists due to the design of the system to stop loss of pool level involving siphon breakers. The pumps may eventually actually trip if left running with no flow, when they seize and trip on overload (Parent question), but this is not considered an "automatic trip".

D - CORRECT: Given the leak location between SFP Cooling Pumps discharge and the SFP Cooling Heat Exchangers, and the fact that the Downstream Check Valves, 2-RW-8 and 2-RW-10, are functioning as designed, any back-leakage from the pool through the discharge line would be stopped, causing the suction line siphon breaker to stop the lowering of pool level.

References

LP SFP-00-C, PPT, R5C2, Pg. 15

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 033 033 (SF8 SFPCS) Spent Fuel Pool Cooling

Question #: **62** **Question ID: 55162** **Rev. 6** ☐ **Student Handout?** ☑ **Lower Order?**

Last Edited: 6/11/2018 3:05:10 PM ☑ **RO** ☐ **SRO** **Origin: Mod** ☐ **Past NRC Exam?**

Number A1.01 **RO** 2.7 **SRO** 3.3 **CFR Link** (CFR: 41.5 / 45.5)

A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Spent Fuel Pool Cooling System operating the controls including: Spent fuel pool water level

Question #: **N/A** **Question ID: 55162** **Rev. 5** ☐ Student Handout? ☐ Lower Order?
☑ **RO** ☐ **SRO** **Origin: Parent** ☑ Past NRC Exam?

PARENT

The SFP Cooling system is in a normal lineup, when the following Control Room overhead annunciators are received:

- SFP CLG PUMP SUCTION FLOW LO
- SFP LEVEL LO
- SFP PUMPS OVERLOAD/TRIP

The cause of the alarms is determined to be a leak between the SFPC Pumps and the SFPC heat exchangers (assume that check valves operate with no leakage).

With no operator action, SFP level will lower until water level reaches:

☐ **A** the bottom of the low suction

☐ **B** 12 ‘6’’ above the storage racks

☑ **C** the suction line siphon breaker

☐ **D** the discharge line siphon breaker

Question Misc. Info: MP2*LOIT*2502 [033 SFP-01-C 7324] (2/5/2001) 2305, SFP, NLIT, NRC-2014

Justification

A - Incorrect: The low or “deep” suction is lower than the suction line siphon break and therefore will not be uncovered.

B - Incorrect: The fuel pool will not drain to this level. Per the SFPC lesson plan, the pool is designed and maintained to prevent inadvertent draining of the pool below a level of 22’6”.

C - CORRECT: The leak is on the suction line and will be stopped by the suction line siphon breaker. The question stem states the examinee is to assume check valves operate with no leakage.

D - Incorrect: The question stem states the examinee is to assume check valves operate with no leakage. Given the leak location between SFP Cooling Pumps discharge and the SFP Cooling Heat Exchangers, the Downstream Check Valves 2-RW-8 and 2-RW-10 would prevent any back-leakage from the pool through the discharge line, without reliance on the discharge line siphon breaker.

Question References not yet listed.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 033 Spent Fuel Pool Cooling System (SFPCS)

Number K4.01 **RO** 2.9 **SRO** 3.2 **CFR Link** (CFR: 41.7)

Knowledge of design feature(s) and/or interlock(s) which provide for the following: Maintenance of spent fuel level

Question #: **63**	***Question ID:*** ***2018026***	**Rev.** **0**	☐ **Student Handout?**	☑ **Lower Order?**
Last Edited: 6/18/2018 7:11:05 PM	☑ ***RO*** ☐ ***SRO***		***Origin:*** ***Mod***	☐ **Past NRC Exam?**

The plant has just tripped from 100% power due to a loss of DV-10. The number 2 main turbine stop valve did not close on the trip.

What is the expected response of the main turbine / generator systems?

☐ **A** Only the number 1 main steam isolation valve (MSIV) will close and the main turbine control valves will prevent steam from reaching the turbine.

☑ **B** Both Main Steam Isolation Valves (MSIV) will close preventing steam from reaching the turbine.

☐ **C** The main turbine / generator bearings will have to be supplied from the emergency bearing oil pump during coastdown.

☐ **D** Hydrogen from the main generator will be automatically vented to atmosphere on the turbine trip.

Question Misc. Info: MP2*ILT 2525, MTC,

Justification

A - INCORRECT: Only the number 1 main steam isolation valve (MSIV) will close and the main turbine control valves will prevent steam from reaching the turbine is not correct. Both MSIVs will close on a loss of DV-10. There are two solenoids in the air path to the MSIV air actuator. One is powered from DV-10 and the other from DV-20. Loss of either vital DC bus will close the air supply and vent air from the actuator causing the MSIVs to close.
PLAUSIBLE: It is reasonable for an examinee to think that there is facility separation and independence and that an MSIV would only be affected by one facility of vital power. Single power supplies to components is normal. Such as the loss of DV-10 would only fail open one auxiliary feedwater control valve.

B - CORRECT: Both main steam isolation valves (MSIV) will close preventing steam from reaching the turbine is correct. The lost of DV-10 closes both MSIVs.

C - INCORRECT: The main turbine / generator bearings will have to be supplied from the emergency bearing oil pump during coastdown is not correct. The turning gear oil pump is the normal supply for turbine bearing lube oil when the turbine is coasting down. The turning gear oil pump is powered from B62. B62 will be powered on a plant trip with the loss of DV-10. The loss of DV-10 will prevent the transfer of the RSST to 24C/24A but will not affect the RSST transfer to 24D/24B. Bus 24D feeds 22F which feeds B62. The emergency bearing oil pump would not be needed and would not be expected to be operating.
PLAUSIBLE: The lost of DV-10 will result in some non-vital power that supports turbine systems being lost. The examinee will understand that the majority of turbine support systems are powered from non-vital power. The turning gear oil pump and lift pumps are powered from vital power which is an exception. The examinee could also think that the emergency bearing oil pump starts on all turbine trips as a backup.

D - INCORRECT: Hydrogen from the main generator will be automatically vented to atmosphere on the turbine trip is not correct. The lost of DV-10 will result in the lost of non-vital bus B-12 which powers the main seal oil pump, recirculating oil pump, and seal oil vacuum pump. These are the main pumps that normally maintain hydrogen in the generator. With the loss of the main seal oil pump the emergency seal oil pump will start and maintain seal oil to the main generator preventing hydrogen from escaping the generator.
PLAUSIBLE: The examinee should understand that most turbine/generator support systems are powered from non-vital power and that some non-vital power was lost on the trip with a lost of DV-10. It is also reasonable to think that hydrogen is only required in the main generator when the plant is operating. And that automatically venting all the hydrogen out of the main generator to atmosphere, when seal oil pumps were lost, would be safer than having hydrogen leak out slowly into the building.

References

EOP 2525, Standard Post Trip Actions
AOP 2506A, Loss of Vital 125 VDC Instrument Panel DV10
HSO-00-C, Lesson Text, Hydrogen Seal Oil System
MSS-00-C, Lesson Text, Main Steam System
MT-00-C, Lesson Text, Main Turbine

NO Comments or Question Modification History at this time.

Question #: **63** **Question ID: 2018026** **Rev. 0** ☐ **Student Handout?** ☑ **Lower Order?**

Last Edited: 6/18/2018 7:11:05 PM ☑ **RO** ☐ **SRO** **Origin: Mod** ☐ **Past NRC Exam?**

NRC K/A System/E/A **System** 045 045 (SF 4S MTG) Main Turbine Generator

Number A3.11 **RO** 2.6* **SRO** 2.9* **CFR Link** (CFR: 41/7 / 45.5)

A3.11 Ability to monitor automatic operation of the MT/G system, including: Generator trip

Question #: **N/A** **Question ID:** **56620** **Rev.** 1 ☐ Student Handout? ☐ Lower Order?

☑ **RO** ☐ **SRO** **Origin:** **Parent** ☐ Past NRC Exam?

PARENT

A reactor trip has occurred from 100% power. While performing EOP 2525, "Standard Post Trip Actions", the BOP notices that one Main Turbine Stop Valve is still open and the 8T and 9T are still closed. All other components are performing as designed per the existing conditions.

Which of the following are the specific actions required and what is the basis for the actions?

☐ **A** First close both MSIV's and trip the 8T and 9T breakers, then verify megawatts at zero and turbine speed lowering.
Ensures the Main Turbine does not overspeed.

☑ **B** First check the Turbine tripped by all Control Valves closed, megawatts at zero and turbine speed lowering, then verifying the 8T and 9T are open.
Ensures the Main Turbine does not overspeed.

☐ **C** First close both MSIV's and trip the 8T and 9T breakers, then verify megawatts at zero and turbine speed lowering.
Ensures bus transfer to the RSST is not delayed.

☐ **D** First check the Turbine tripped by all Control Valves closed, megawatts at zero and turbine speed lowering, then verifying the 8T and 9T are open.
Ensures bus transfer to the RSST is not delayed.

Question Misc. Info: MP2*LOIT*, MB-00381, E25-01-C, MB-05425

Justification

The action in answer "B" is addressed in EOP 2525. The reason is in the EOP Basis document.

References

EOP 2525, R28, C0, ST. 2 and Tech. Guide for EOP 2525, R27, St. 2

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 045 045 (SF 4S MTG) Main Turbine Generator

Number A3.11 **RO** 2.6* **SRO** 2.9* **CFR Link** (CFR: 41/7 / 45.5)

A3.11 Ability to monitor automatic operation of the MT/G system, including: Generator trip

Question #: **64** **Question ID:** **62039** Rev. 5 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/14/2018 10:57:49 AM ☑ RO ☐ SRO **Origin:** **Bank** ☐ Past NRC Exam?

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

Which of the following events would NOT affect Main Condenser vacuum?

☐ **A** Lowering steam pressure to the Steam Jet Air Ejectors.

☑ **B** The Gland Seal regulator fails closed.

☐ **C** Condensate depression is being reduced.

☐ **D** Loss of main condenser vacuum breaker water seal.

Question Misc. Info: MP2*ILT MB-00339, CAR

Justification

A - WRONG; The SJAEs are very sensitive to changes in their steam supply pressure. The procedure cautions the pressure should be maintained at approximately 210 psig. Plant experience has shown that if it drops only slightly below 200 psig, they will substantially degrade in performance and condenser vacuum will degrade.
PLAUSIBLE; The student may recall the normal pressure range to start the SJAEs is 200 - 220 psig, and not know they are so sensitive to a drop in pressure below this.

B - CORRECT; As the plant power is driven up during a power ascension, more and more steam leaks out of the High Pressure Turbine shaft glands and effectively pressurizes the Gland Sealing Steam system. By the time the plant is at 100% power, steam supply from the Gland Seal Regulator is no longer needed and it would be closed.

C - WRONG; When seasonal changes can cause Circ. Water temperature to be so low that it begins to lower plant efficiency. When this happens, condensate hotwell temperature is raised by lowering Circ. Water flow and intentionally degrading condenser vacuum.
PLAUSIBLE; The student may not know what or how condensate depression is managed.

D - WRONG; Although the vacuum breaker valve is always closed at power and the valve is maintained as leak-free as possible, if the water seal evaporates completely air in leakage is often the result.
PLAUSIBLE; Student may believe the water seal is only there to indicate that the valve seat is degrading and in need of repair (and is the source of any present air in leakage), but is not normally required to maintain a valve seal.

References

OP 2323D, R10C0, Discussion.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 055 055 (SF4S CARS) Condenser Air Removal

Number K3.01 **RO** 2.5 **SRO** 2.7 **CFR Link** (CFR: 41.7 / 45.6)

K3.01 Knowledge of the effect that a loss or malfunction of the CARS will have on the following: Main condenser

Question #: **65** **Question ID: 8066113** **Rev. 1** ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/14/2018 8:28:03 AM ☑ **RO** ☐ **SRO** **Origin: Bank** ☐ Past NRC Exam?

"A" Clean Waste Monitor Tank (CWMT) Discharge has just been started when the following occurs:

- CLEAN WASTE PANEL TROUBLE alarm is received on C-06/7.
- "RM-9049 LOSS OF FLOW" annunciator is in alarm on the Clean Waste Panel.
- The sample pump is not running and the discharge has automatically secured.
- I&C reports the rad monitor sample pump must be repaired or replaced.

Shift turnover is about to start so the US directs the discharge permit be closed out and reevaluated on the next shift.

Which of the following actions must be performed, prior to obtaining a new discharge permit, to restart the discharge?

☐ **A** The discharge flow rate must lowered to match sample flow.

☐ **B** I&C must repair or replace the failed rad monitor sample pump.

☐ **C** The clean liquid radwaste discharge final filter must be changed.

☑ **D** Chemistry must obtain two independent samples of the tank.

Question Misc. Info: MP2*NLO NLIT, PIOPS, 2317A, NRC-2008

Justification

A - WRONG; Sample flow is a slip-stream off of discharge flow that is driven by a sample pump, and is not significantly affected by discharge flow.
PLAUSIBLE; Student may believe this discharge evolution is similar to a waste gas discharge, which must often be started slowly to prevent a spike in rad monitor indication from securing the discharge.

B - WRONG; The discharge may be accomplished with a failed rad. Monitor if 2 samples are analyzed by chemistry.
PLAUSIBLE; Student may be taking a strict interpretation of the administrative requirements to secure a discharge on a rad. monitor failure.

C - WRONG; This would cause a reduction in discharge flow, not sample flow.
PLAUSIBLE; Student may believe the sample slip stream could be effected by a clogged discharge filter because it would result in a decrease in the discharge flow.

D - CORRECT; The RM-9069 Loss of Flow annunciator is caused by low sample pump flow (due to a failed sample pump). The loss of indicated sample flow to the Rad Monitor will result in an inoperable Rad Monitor. ARP 2593C-023 requires the operator to refer to REMODCM 1.V.C.1 to allow continuation of the discharge with an inoperable Rad Monitor. In order to perform a discharge with an inoperable Rad Monitor, Chemistry must analyze two independent samples.

References

SP 2617D, R1C0, St. 4.1.9a and 4.1.17

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 068 068 (SF9 LRS) Liquid Radwaste

Number K6.10 **RO** 2.5 **SRO** 2.9 **CFR Link** (CFR: 41.7 / 45.7)

K6.10 Knowledge of the effect of a loss or malfunction on the following will have on the Liquid Radwaste System : Radiation monitors

Question #: **66** **Question ID: 2014034** **Rev. 1** ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/18/2018 7:20:46 PM ☑ **RO** ☐ **SRO** **Origin: Mod** ☐ Past NRC Exam?

Reactivity Management procedure, OP-AA-300 requires operators to ensure power is reduced to maintain margin to less than 100.0 percent before conducting evolutions that are known to add positive reactivity.

Which one of the following evolutions would add positive reactivity as stated in OP-AA-300?

☑ **A** Raising the steam generator blowdown flow rate

☐ **B** Forcing pressurizer spray flow

☐ **C** Raising the letdown temperature

☐ **D** Running a motor driven auxiliary feedwater pump on recirc

Question Misc. Info: MP2*LOIT NRC-2014

Justification

OP-AA-300 states; **ENSURE** power is reduced to maintain margin to less than 100.0 percent before conducting evolutions that are known to add positive reactivity. For example:
- Operating steam-driven auxiliary feed water pump turbines
- Lowering letdown temperature
- Increasing steam generator blowdown flow rate
- Flowing auxiliary feed to the steam generators
- Increasing auxiliary steam supply flow
- Changing feed water conditions
- Condensate polisher evolutions

A - CORRECT: Increasing steam generator blowdown flow rate will add positive reactivity because it increases RCS heat removal. Increased RCS heat removal will lower RCS temperature which adds positive reactivity due to the negative moderator temperature coefficient.

B - INCORRECT: Forcing pressurizer sprays will not add postive reactivity if performed properly and it is not one of the listed in OP-AA-300.
PLAUSIBLE: The examinee should understand that raising pressurizer pressure will add positive reactivity as a result of the density increase (less voids) but pressure is not intended to be raised when forcing pressurizer sprays. Pressure could rise some starting to force sprays. The operator would be expected to adjust back to initial value.

C - INCORRECT: Raising the letdown temperature adds negative reactivity. When letdown temperature is raised boron is released from resin.
PLAUSIBLE: Applicant may confuse the reactivity effect of raising letdown temperature with lowering letdown temperature which DOES add positive reactivity and requires a power reduction beforehand in accordance with OP-AP-300 Reactivity Management.

D - INCORRECT: Running a motor driven auxiliary feedwater pump on recirc will not add positive reactivity because it is not feeding the steam generator (adding cold feedwater to the steam generator) nor is it using steam to operate it like the turbine driven auxiliary feedwater pump does.
PLAUSIBLE: Flowing auxiliary feedwater to the steam generators is listed as an evolution that is known to add positive reactivity. The applicant may understand that auxiliary feedwater is listed and not understand that it requires flow to the steam generators to add positive reactivity.

References

OP-AA-300, Reactivity Management

NO Comments or Question Modification History at this time.

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

Question #: **66** **Question ID: 2014034** **Rev. 1** ☐ **Student Handout?** ☑ **Lower Order?**

Last Edited: 6/18/2018 7:20:46 PM ☑ **RO** ☐ **SRO** **Origin: Mod** ☐ **Past NRC Exam?**

Generic K/A Selected

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

Number 2.1.1 **RO** 3.8 **SRO** 4.2 **CFR Link** (CFR: 41.10 / 45.13)

G2.1.1 Knowledge of conduct of operations requirements.

Question #: **N/A** **Question ID: 2014034** Rev. 0 ☐ Student Handout? ☐ Lower Order?

☑ **RO** ☐ **SRO** **Origin: Parent** ☑ Past NRC Exam?

PARENT

QUESTION 67
The plant is operating in MODE 1 at full power.

In this plant condition, for which ONE of the following evolutions does OP-AP-300, Reactivity Management, require the crew to ensure power is reduced to maintain margin to less than 100.0 percent before conducting the evolution?

☐ **A** Lowering steam generator blowdown rate

☐ **B** Stopping a CW pump and isolating the associated CW pump's waterbox

☐ **C** Increasing the temperature of letdown

☑ **D** Overspeed testing of the AFP turbine following trip mechanism repair

Question Misc. Info: MP2*LOIT NRC-2014

Justification

A - Incorrect: Lowering steam generator blowdown rate does not add positive reactivity.
Plausible: Applicant may confuse this with RAISING steam generator blowdown rate which adds positive reactivity.

B - Incorrect: Stopping a circ water pump and isolating a water box does not add positive reactivity.
Plausible: OP2325A Circulating Water System directs lowering power if condenser backpressure is hard to maintain when stopping a circulating water pump and cross-tying water boxes.

C - Incorrect: Raising letdown temperature adds negative reactivity (boron release from resin).
Plausible: Applicant may confuse reactivity effect of raising letdown temperature with lowering letdown temperature which DOES add positive reactivity and requires a power reduction beforehand in accordance with OP-AP-300 Reactivity Management.

D - CORRECT: The overspeed trip test of the AFW pump turbine adds positive reactivity. Steam is admitted to the turbine to perform this test when in MODES 1 thru 4, in accordance with SP-2660. OP-AP-300 Attachment 2 states, "ENSURE power is reduced to maintain margin to less than 100.0 percent before conducting evolutions that are known to add positive reactivity….for example…operating steam-driven auxiliary feed water pump turbines."

Question References not yet listed.

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.1.37 **RO** 4.3 **SRO** 4.6 **CFR Link** (CFR: 41.1 / 43.6 / 45.6)

Knowledge of procedures, guidelines, or limitations associated with reactivity management.

Question #: **67** **Question ID: 2018025** Rev. 0 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/14/2018 7:42:09 AM ☑ RO ☐ SRO **Origin: New** ☐ Past NRC Exam?

The plant is operating at 100% power, steady state.
Four (4) Condensate Polishing Demineralizers are in service, one being in Amine Form.
2-CNM-2, "COND DEMIN BYPASS" is throttled open.
Chemistry is in the process of regenerating two condensate polishing demineralizers.

Suddenly, a large Condenser Tube Leak develops in the "A" Waterbox.

Prior to changing plant power, which of the following operations should be immediately terminated, as required by AOP 2516, "Condenser Tube Leak"?

☑ **A** Terminate any liquid radwaste discharges in progress.

☐ **B** Terminate condensate slip-stream flow around CPF.

☐ **C** Terminate sodium hypochlorite injection to CW and SW pumps.

☐ **D** Terminate flow through the Amine Form demineralizer.

Question Misc. Info: MP2*LOIT, Condensate, Demin, amine, 2516, 2319, NRC-2008

Justification

A - CORRECT; Immediately after notifying chemistry of the condenser tube leak, the AOP directs to terminate any liquid radwaste discharge in progress and any Circ Water bay shocking. This is a preemptive action based on the fact that Circ. Water flow will probably be lowered in an attempt to isolate the condenser tube leak.

B - WRONG; Slip-stream flow cannot be immediately terminated due to having only 4 demineralizers in service at 100% power.
PLAUSIBLE; Student may remember that this is a priority of the AOP and that with a third condensate pump now in standby at 100% power, immediately closing the bypass valve could be tolerated without tripping the plant on loss of feed flow.

C - WRONG; The AOP requires Circ. Water bay shocking be terminated (NaHClO injection to CW), but not injection to Service Water, which occurs 24/7.
PLAUSIBLE; Student may remember the procedure requires NaHClO injection be stopped, but not where.

D - WRONG; Eventually, in the AOP followup actions, all Amine Form demineralizers are usually removed from service, but not until directed by Chemistry.
PLAUSIBLE; Student may recall it is desired to remove these demineralizers from service during a condenser tube leak.

References

AOP 2516, Condenser Tube Leak, R2C0, Initial Actions.

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.1.34 **RO** 2.7 **SRO** 3.5 **CFR Link** (CFR: 41.10 / 43.5 / 45.12)
G2.1.34 Knowledge of primary and secondary plant chemistry limits.

Question #: **68** **Question ID:** **53860** **Rev.** 1 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/7/2018 3:19:46 PM ☑ **RO** ☐ **SRO** **Origin:** **Bank** ☐ Past NRC Exam?

While moving fuel during a refueling outage, the following conditions are present:

- The outage is 24 hours behind the original schedule.
- MSIVs are closed.
- #1 S/G is to be drained for maintenance.
- #1 Atm Dump Valve is opened to provide a vent path.

Outage control has just learned that Maintenance has just removed the secondary manway cover on # 1 S/G, exactly as planned on the original schedule.

Which one of the following statements is correct for fuel movement?

☐ **A** It must be suspended until HP has approved the change in Containment conditions.

☐ **B** It can continue provided the appropriate TSAS has been entered.

☑ **C** It must be suspended until Containment Integrity is re-established.

☐ **D** It may continue provided Containment atmosphere is maintained at a negative pressure.

Question Misc. Info: MP2*LORT*4943 [034 REF-01-C 4982] (12/1/97) C94407 MP2, LCO 3.9.4, 2303, 2614C, FH

Justification

A - WRONG; HP is not responsible for maintaining CTMT Integrity or ensuring all requirements are met for fuel movement.
PLAUSIBLE; Student may believe that because HP monitors all outage activities in CTMT (including fuel movement), and has the authority to stop any activity that may increase the risk for radiation exposure, that they would also be involved in approving fuel movement activities.

B - WRONG; The outage schedule would not have planned to violate the CTMT integrity TS LCO 3.9.4.c, as it states to immediately suspend all CORE ALTERATIONS or movement of irradiated fuel in the containment.
PLAUSIBLE; Student may recognize that the plant routinely cycles through several Tech. Specs. And TRM conditions as various components are repaired and tested. With all outage activities carefully planned, this may simply be another area where a TSAS is temporarily entered until the planned activities are completed.

C - CORRECT; Opening the ADV with a manway cover removed violates Containment Integrity. The applicable TSAS LCO 3.9.4.c states immediately suspend all CORE ALTERATIONS or movement of irradiated fuel in the containment. A closure plan could also be established that allows the ADV to be opened as long as it would be closed by a responsible individual designated for the purpose if a fuel handling accident were to occur. But fuel handling would have to be stopped while the closure plan was prepared and authorized.

D - WRONG; The CTMT must be maintained at a negative pressure if fuel movement is in progress. However, this assumes CTMT integrity can be quickly established by closing the access door.
PLAUSIBLE; Student should remember that CTMT is usually maintained at a negative, and may believe this is to allow for evolutions that temporarily violate CTMT Integrity.

References

TSAS 3.9.4c.1

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.1.36 **RO** 3.0 **SRO** 4.1 **CFR Link** (CFR: 41.10 / 43.6 / 45.7)

G2.1.36 Knowledge of procedures and limitations involved in core alterations.

Question #: **69** **Question ID:** **1154135** **Rev.** 2 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/15/2018 2:15:18 PM ☑ **RO** ☐ **SRO** **Origin:** **Mod** ☐ Past NRC Exam?

A valve lineup is being performed on a system that requires Independent Verification.
The valve lineup contains a valve that is required to be throttled open two full turns.
Permission has been given to move the valve as required to verify its position.

What are the actions necessary to verify the position of this throttled open valve, in accordance with PI-AA-500?

☐ **A** One operator will go out ALONE and fully close the valve, then reopen it two full turns. Then, a second operator will go out ALONE and verify the valve is properly positioned by visual observation of the valve stem.

☐ **B** One operator will go out ACCOMPANIED by the second operator. One of the operators will fully open the valve, then close the valve the same number of turns, while the second operator observes the actions.

☐ **C** One operator will go out ACCOMPANIED by the second operator. One of the operators will fully close the valve, then reopen it two full turns, while the second operator observes the actions. Then, a third operator will go out ALONE and verify the valve is properly positioned by visual observation of the valve stem.

☑ **D** One operator will go out ACCOMPANIED by the second operator. One of the operators will fully close the valve, then reopen it two full turns, while the second operator observes the actions.

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

Justification

A - WRONG; Verification using the valve stem is only allowed if the valve is "sealed" in the known, throttled position.
PLAUSIBLE; This is the acceptable method if the throttle valve was "sealed" in position.

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 to some new desired position based on observation of a system parameter, especially if totally stopping flow was unacceptable.

C - WRONG; This is not the actions specified in PI-AA-500 for verification of a throttled valve.
PLAUSIBLE; The method described is a combination of what is done for a locked throttle valve (Concurrent Verification to check position) that requires Independent Verification (required to check valve is locked).

D - 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 and how it should be done.

References

PI-AA-500, R3C0, Att. 2, St. "c"

NO Comments or Question Modification History at this time.

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)

G2.2.14 Knowledge of the process for controlling equipment configuration or status.

Question #: **N/A** **Question ID:** **1154135** **Rev.** 1 ☐ Student Handout? ☐ Lower Order?

☑ **RO** ☐ **SRO** **Origin:** **Parent** ☐ Past NRC Exam?

PARENT

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.

Question References not yet listed.

NO Comments or Question Modification History at this time.

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)

G2.2.14 Knowledge of the process for controlling equipment configuration or status.

Question #: **70** **Question ID:** **64620** **Rev.** 5 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/14/2018 8:30:06 AM ☑ **RO** ☐ **SRO** **Origin:** **Bank** ☐ Past NRC Exam?

While scanning the annunciators, you notice a "Blue Dot" on one of the annunciator windows.

What is the significance of the blue dot?

☐ **A** A compensatory action is triggered by the presence of an alarm and alarm response can be delayed during a transient.

☐ **B** The annunciator is triggered by a local panel alarm and control room operators do not need to refer to the alarm response.

☑ **C** The associated alarm logic card has been removed due to nuisance alarms that may or may not require additional actions.

☐ **D** The annunciator can be triggered by multiple inputs and one or more of these inputs are known to be out of service.

Question Misc. Info: MP2 LORT LOIT ANN 2387A, Significance of Blue Dot on Annunciator window, NRC-2008

Justification

A - WRONG; This annunciator would be indentified by a "border" placed around the annunciator window.
PLAUSIBLE; Student may remember any alarm that refers to compensatory actions that can be delayed during a transient will have the annunciator window marked.

B - WRONG; This annunciator would be indentified by a "green triangle" in the lower right corner of the window.
PLAUSIBLE; Student may remember local panel alarms have a special designation on the window.

C - CORRECT; Per OP-2387A; A "blue dot" is placed on an annunciator window to indicate that it is a hanging or nuisance alarm that was taken out of service. An annunciator is considered out of service when the associated alarm (logic) card is removed.

D - WRONG; This condition would require an Orange Dot on the annunciator window.
PLAUSIBLE; Student may remember the dot signifies the annunciator will not alarm for a given condition, but not that the Blue Dot means it will not alarm for ANY condition.

References

OP 2387A, Annunciator System Operation and Control

NO Comments or Question Modification History at this time.

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 2.2.43 **RO** 3.0 **SRO** 3.3 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

G2.2.43 Knowledge of the process used to track inoperable alarms.

Question #: **71** **Question ID: 1000055** Rev. 3 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/17/2018 11:28:53 AM ☑ **RO** ☐ **SRO** **Origin: Bank** ☐ Past NRC Exam?

"A" Clean Waste Monitor Tank is in the process of being discharged.

Which of the following are required to be checked by SP-2617D, "Clean Radioactive Liquid Waste Discharges" procedure approximately 15 minutes after the discharge is started?

☑ **A** Discharge radiation monitor channel check and system discharge flow rate

☐ **B** Discharge radiation monitor channel check and radiation monitor sample flow rate

☐ **C** Clean Waste final filter D/P and system discharge flow rate

☐ **D** Clean Waste final filter D/P and radiation monitor sample flow rate

Question Misc. Info: MP2 LOIT, ADM-02-J, MB-4815, CFR55.43.b.2

Justification

A - CORRECT: Discharge radiation monitor reading and system discharge flow rate is correct. SP-2617D, "Clean Radioactive Liquid Waste Discharges" specifies that when 15 minutes have elapsed that the discharge radiation monitor (RM-9049 indication and system discharge flow (FR-9118) indication be checked. These two checks are important because the radiation monitor is used to automatically terminate the discharge and requires a CHANNEL CHECK. The system discharge flow rate is used to determine the amount discharged as well to maintain the discharge within the permit required maximum approved discharge flow rate.

B - INCORRECT: Discharge radiation monitor reading and radiation monitor sample flow rate is not correct. The discharge radiation monitor reading is correct but the radiation monitor sample flow is not correct. Radiation monitor sample flow rate is not checked by the procedure after 15 minutes.
PLAUSIBLE: Sample flow is important to the discharge process and the ability to monitor and control the release. Sample flow is initially setup between a specified range to obtain a represent radiation reading through the discharge radiation monitor. Improper sample flow will affect the discharge radiation monitor reading. If the sample pump stops the discharge will automatically terminate.

C - INCORRECT: Clean Waste final filter D/P and system discharge flow rate is not correct. System discharge flow rate is correct. Clean Waste final filter D/P is not correct. The procedure requires the discharge be terminated if D/P reaches 40 psid but it is not required to be checked after 15 minutes of the start of the discharge.
PLAUSIBLE: Clean Waste final filter D/P is required to be monitored and is important to monitor because it would affect discharge flow if the D/P gets to high. Discharge filter D/P is also recorded after the discharge is complete.

D - INCORRECT: Clean Waste final filter D/P and radiation monitor sample flow rate are not correct. Both are incorrect. The procedure requires the discharge be terminated if D/P reaches 40 psid but it is not required to be checked after 15 minutes of the start of the discharge. Radiation monitor sample is not checked by the procedure after 15 minutes.
PLAUSIBLE: Sample flow is important to the discharge process and the ability to monitor and control the release. Clean Waste final filter D/P is required to be monitored and is important to monitor because it would affect discharge flow if the D/P gets to high.

References

SP 2617D, R1C0, St. 4.1.41

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.3 Radiation Control

Generic K/A Selected

NRC K/A Generic **System** 2.3 Radiation Control

Number 2.3.11 **RO** 3.8 **SRO** 4.3 **CFR Link** (CFR: 41.11 / 43.4 / 45.10)

G2.3.11 Ability to control radiation releases.

Question #: **72** | **Question ID: 2018030** | **Rev. 0** | ☐ Student Handout? | ☐ Lower Order?

Last Edited: 6/14/2018 8:31:54 AM | ☑ **RO** | ☐ **SRO** | **Origin: Mod** | ☐ Past NRC Exam?

The plant is operating in MODE 5 with Excess Letdown flow in operation, in accordance with OP 2304F, "CVCS Operation While in Cold Shutdown".
Then, plant chemistry reports indications of increased RCS activity due to an unexpected crud burst.
The SM directs Excess Letdown operations be secured and Additional Purification flow be started to clean up the crud.

Which of the following areas will no longer have radioactive RCS water flowing through them, based on the change in system alignment from Excess Letdown to Additional Purification flow?

☑ **A** The three Charging Pump Rooms

☐ **B** The Letdown Post-Filter L-20 blockhouse.

☐ **C** The area around 2-SI-709, “SDC Suction Header Isolation”

☐ **D** LPSI Loop Injection Valve Area.

Question Misc. Info: MP2*LOIT, ALARA, CVCS, 2304, 2207, NRC-2011, Audit-2018

Justification

A - CORRECT; Excess Letdown does go through the Charging Pumps, however Additional Purification does NOT. Therefore, the charging pump area will see a drop in radiation levels.
PLAUSIBLE: The student may confuse Excess Letdown with Additional Purification.

B - WRONG; Letdown Flow will be diverted from the VCT and piped through 2-CH-024, SDC Suction to CVCS. However, CH-024 taps off downstream of L-20, therefore, radiation levels going through filter L-20 (i.e.; in the blockhouse) are unchanged.
PLAUSIBLE: The student could remember that Additional Purification bypasses the VCT and believe it splits off and returns to the LPSI Pump suctions before L-20.

C - WRONG; SI-709 is still in the flow path for Additional Purification, but will not see the filtered RCS. Therefore, radiation levels here will not change.
PLAUSIBLE: The student may believe the letdown flow return to SDC taps in upstream of SI-709.

D - WRONG; The flow through the injection valves has not changed, although they will see <u>some</u> of the filtered RCS water passing through them. However, these valves are located in the same area as SDC suction from the RCS, which is initially seeing the unfiltered water.
PLAUSIBLE: Student may feel rad levels will go down based on cleaner water passing through the injection valves.

References

LP CVC-00-C, Figs. 14 and 15

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.3 Radiation Control

Generic K/A Selected

NRC K/A Generic **System** 2.3 Radiation Control

Number 2.3.14 **RO** 3.4 **SRO** 3.8 **CFR Link** (CFR: 41.12 / 43.4 / 45.10)

G2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Question #: **N/A** **Question ID: 1100057** **Rev. 1** ☐ Student Handout? ☐ Lower Order?

☑ **RO** ☐ **SRO** **Origin: Parent** ☑ Past NRC Exam?

PARENT

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.

Question References not yet listed.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.3 Radiation Control

Generic K/A Selected

NRC K/A Generic **System** 2.3 Radiation Control

Number 2.3.14 **RO** 3.4 **SRO** 3.8 **CFR Link** (CFR: 41.12 / 43.4 / 45.10)

G2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Question #: **73** **Question ID:** **78980** **Rev.** 6 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/14/2018 10:37:49 AM ☑ **RO** ☐ **SRO** **Origin:** **Bank** ☐ Past NRC Exam?

On the Accident Monitoring Panel, C-101, next to the Unit 2 stack Gaseous and Particulate High Range Radiation Monitor (Kaman), RM-8168, there is a handswitch labeled "ENABLED/DISABLE".

If RM-8168 were to fail, which of the following describes the effect of placing this handswitch in its "DISABLE" position?

☐ **A** Prevents rad monitor input to the MIDAS offsite dose calculation program.

☑ **B** Prevents an automatic purge of the MP2 stack rad monitor, RM-8132B.

☐ **C** Prevented triggering of the "RAD HI-HI/FAIL" annunciator on C-06/7.

☐ **D** Prevents the rad monitor from tripping all Main Exhaust Fans.

Question Misc. Info: MP2*LOIT/LORT, RMS-01-C, RM, 2383, 2314A

Justification

A - WRONG; The ENABLE/DISABLE switch has no effect on the output of the Kaman Rad Monitor to the MIDAS program.
PLAUSIBLE; The student may remember that the Kaman Rad Monitor is a high range radmonitor located on the Accident Monitoring Panel and is part of the post-accident equipment used to monitor for potential high radiation releases due to massive failures of all three barriers (Fuel, RCS and CTMT). It is logical to assume that the switch exists to prevent a failed signal from being used for this purpose.

B - CORRECT: When in "ENABLED" and the Kaman (RM-8168), reaches its Hi Rad alarm setpoint, an automatic purge of RM-8132B occurs. The "DISABLE" position is for I&C testing to prevent auto purge of RM-8132B.

C - WRONG; Unlike the vast majority of radmonitors in the plant, the Kaman radmonitor does not input to the control room radmonitor panel RC-14. It has its own Hi Rad/Fail annunciator on C-06/7.
PLAUSIBLE; The student may remember that disabling alarms on RC-14 will clear the "RAD HI-HI/FAIL" alarm on C-06/7.

D - WRONG; An alarm on the Kaman will purge the Stack Radmonitor, not the other way around.
PLAUSIBLE; The student may remember that the Kaman is used for Emergency Plan Classification and therefore logical that it is desired to allow it to be initially purged of contaminants and once clear, prevented from being purged for classification determination.

References

ARP 2590E-061, R2C0, VENT STACK RADMONITOR HI/FAIL

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.3 Radiation Control

Generic K/A Selected

NRC K/A Generic **System** 2.3 Radiation Control

Number 2.3.5 **RO** 2.9 **SRO** 2.9 **CFR Link** (CFR: 41.11 / 41.12 / 43.4 / 45.9)

G2.3.5 Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Question #: **74** **Question ID: 2018029** Rev. 0 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/17/2018 11:29:42 AM ☑ **RO** ☐ **SRO** **Origin: New** ☐ Past NRC Exam?

A reactor trip occurred and the crew is implementing EOP 2525, Standard Post Trip Actions. The following are the current conditions of the plant:

- An Excess Steam Demand event is in progress on steam generator #1 and CET temperatures are raising
- Pressurizer pressure is 1400 psia and SIAS, CIAS, EBFAS have actuated
- Two CEA did not insert and emergency boration was started

The crew is at the RCS Inventory Control step and neither the RO nor BOP has progressed beyond this step. THEN offsite power is lost and all systems respond as designed.

Of the actions and bases listed which is the operator required to performed first AND what is the reason for this action.

☐ **A** Stabilize CET temperature using the least affected steam generator Atmospheric Dump valve, prevent pressurized thermal shock

☐ **B** Perform the critical task to isolate feedwater to the #1 steam generator, prevent containment pressure from exceeding design limits

☐ **C** Check vital and non-vital busses energized, the loss of offsite power could affect the availability of electrical busses

☑ **D** Check reactivity control safety function safety, the loss of offsite power could have affected emergency boration.

Question Misc. Info: MP2*ILT 2525, implementation, Bases, prioritization

Justification

A - INCORRECT: Stabilize CET temperature using the least affected steam generator Atmospheric Dump valve will not be performed first after the loss of offsite power. OP 2260, the Unit 2 EOP User's Guide, specifies that the US shall perform all portions of SPTAs in the order written and that the RO/BOP shall perform all applicable steps in SPTAs for their watch station in order. The stabilize CET temperature step is deeper into the procedure than the RCS Inventory Control step and the crew has not reached this step. It is correct that this action is performed to prevent pressurized thermal shock.
PLAUSIBLE: The SPTA response not obtained step states that when CET temperature are rising, then operate the ADV for the least affected steam generator to stabilize CET temperature. This is a conditional step and the examinee could reasonable believe that the step is performed when the conditions are met.

B - INCORRECT: Isolating feedwater to the steam generator with the Excess Steam Demand event will not be performed first after the loss of offsite power. Feedwater is isolated to prevent exceeding containment design pressure. OP 2260, the Unit 2 EOP User's Guide, specifies that the US shall perform all portions of SPTAs in the order written and that the RO/BOP shall perform all applicable steps in SPTAs for their watch station in order. This applies to any step including steps that are critical tasks. The isolate feedwater step has not been reached. It is after the RCS Inventory Control step. The critical task requires isolation of feedwater within 30 minutes. This can easily be met implementing the procedure in the order written.
PLAUSIBLE: Isolating feedwater to the steam generator with the Excess Steam Demand event is a critical task. The procedure response not obtained for RCS heat removal directs isolating auxiliary feedwater to the most affected steam generator when steam generator pressure is < 572 psia and an ESDE is in progress. It is reasonable for the examinee to think that whenever this condition occurs that the action is performed.

C - INCORRECT: Checking vital and non-vital busses energized would not be performed first after the loss of offsite power. On a major change, such as a loss of offsite power, the crew must return to step 1 to assess the affect the change had on previous safety functions. It is correct that that the loss of offsite power could affect the availability of electrical busses.
PLAUSIBLE: Procedure guidance directs that if a major change occurs that the US start the procedure again. It is reasonable that they would only go back to the electrical step since the major change was electrical.

D - CORRECT: Check reactivity control safety function after the loss of offsite power would be performed first. It is the highest safety function. The reactivity control safety function could have been affected and emergency boration needs to be verified. OP 2260, the Unit 2 EOP User's Guide, specifies that on a major change, such as a loss of offsite power, the crew must return to step 1.

References

OP 2260, Unit 2 EOP User's Guide
EOP 2525, Standard Post Trip Actions

NO Comments or Question Modification History at this time.

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

Question #: **74** **Question ID: 2018029** **Rev. 0** ☐ **Student Handout?** ☐ **Lower Order?**

Last Edited: 6/17/2018 11:29:42 AM ☑ **RO** ☐ **SRO** **Origin: New** ☐ **Past NRC Exam?**

Generic K/A Selected

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

Number 2.4.23 **RO** 3.4 **SRO** 4.4 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

G2.4.23 Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.

Question #: **75** **Question ID: 153723** **Rev. 1** ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/14/2018 9:17:48 AM ☑ **RO** ☐ **SRO** **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 of the following actions 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** Cross-tie Instrument Air with Unit 3 Service Air.

☐ **D** A battery charger is restored to Facility 2.

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

Justification

A - WRONG; Power must be restored within 4 hours of the reactor shutdown. However, lack of power should not delay feeding the S/Gs because the Turbine Driven AFW pump is assumed available.
PLAUSIBLE; The student may recall that power is required to utilize the electric AFW pumps and that the EOPs expect two pumps to be used to feed the SGs.

B - 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.

C - WRONG; IA restoration is not a TCOA of AOP 2579.
PLAUSIBLE; Student may recall that IA restoration is performed in AOP 2579A, but confuse it with the TCOAs that are required. It is also an action that would be performed as soon as possible.

D - WRONG; A 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; Student may recall that the station batteries have very limited capacity and would be drained faster in an accident situation, therefore, believe that this is a requirement that must be done expeditiously.

References

AOP 2579A, R12C0, Caution preceding Step 3.1

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.4.25 **RO** 3.3 **SRO** 3.7 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of fire protection procedures.

Question #: **76** | ***Question ID:*** ***2018028*** | **Rev.** 0 | ☐ Student Handout? | ☐ Lower Order?

Last Edited: 6/14/2018 8:33:42 AM | ☐ ***RO*** | ☑ ***SRO*** | ***Origin:*** ***New*** | ☐ Past NRC Exam?

The plant has tripped from 100% power due to a failed open PORV, 2-RC-402.
The following conditions have or still exist:

- EOP 2532, Loss Of Coolant Accident, is in progress.
- Facility 1 Block Valve, 2-RC-403 indication: green light energized, red light out.
- Facility 1 PORV, 2-RC-402 indication: green light out, red light energized.
- HPSI Throttle/Stop criteria are now met.
- Pressurizer level is 75% and slowly being lowered using Letdown flow.

Management wants the RCS temperature and pressure reduced to 270°F and 225 psia for placing SDC in service.

Which of the following describe the administrative requirements to reduce the RCS to those conditions?

☐ **A** All but one charging pump must be made incapable of injecting into the RCS.

☐ **B** CVCS must be in service with Pressurizer level reduced to less than 70%.

☑ **C** PORV Block Valve 2-RC-403 must be de-energized in the closed position.

☐ **D** Both of the LT/OP Setpoint Selector switches must be in the "LOW" position.

Question Misc. Info: MP2*ILT 2532, LOCA, TSAS, LTOP, Tech. Spec. 4.3.7

Justification

A - WRONG; Tech. Spec. 3.4.9.3, Overpressure Protection Systems, requires a maximum of one charging pump be capable of injecting into the RCS when Tcold is </= 190°F.
PLAUSIBLE; Student may remember the Tech. Spec. Requirement for LT/OP is ensuring only one charging pump is capable of injecting, but not at what specific temperature it applies.

B - WRONG; The Tech. Spec. For PZR level requires the plant be reduced to Mode 3 in 6 hours (already there) and Mode 4 in the next 4 hours. Therefore it is not a requirement for the conditions.
PLAUSIBLE; Student may realize the PZR being too high means it may not meet its design requirement of preventing a solid condition in the RCS, which would challenge LT/OP even more.

C - CORRECT; Tech. Spec. 3.4.3, Relief Valves, states that with one PORV not capable of being cycled (open & closed) within one hour either fix the PORV or close the associated block valve and then de-energize it. Although this Tech. Spec. is only applicable down to Mode 4 (300°F), if the block valve were allowed to reopen, RCS pressure control would immediately be lost.

D - WRONG; If the LT/OP switch for 2-RC-402 were put in the "LOW" position with the block valve energized (as indicated by its green light lit) the block valve would automatically open and could not be maintained closed.
PLAUSIBLE; Student may remember EOP 2532 directs the switch be placed in LOW if the LT/OP annunciator alarms when RCS Tcold drops to 280°F and </= 375 psia.

References

TS 3.4.3, PROVs and ARP 2590B-043 and 2590B-062.

NO Comments or Question Modification History at this time.

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

Number AA2.05 **RO** 3.9 **SRO** 3.9 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: PORV isolation (block) valve switches and indicators

Question #: **77** **Question ID:** **1100035** Rev. 3 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/14/2018 10:51:15 AM ☐ RO ☑ SRO **Origin:** **Bank** ☐ 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 pressure allowed for "A" and "B" RCP operation.
- "A" and "B" RCPs have 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. Before "A" RBCCW Header flow is restored, 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" RCPSs.

☑ **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 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

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 another RCP 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.

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.

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.

Question #: **77** **Question ID: 1100035** **Rev.** 3 ☐ **Student Handout?** ☐ **Lower Order?**

Last Edited: 6/14/2018 10:51:15 AM ☐ **RO** ☑ **SRO** **Origin: Bank** ☐ **Past NRC Exam?**

PLAUSIBLE; The examinee may believe that starting "C" & "D" RCPs would be acceptable if "A" & "B" were allowed to run.

References

ARP 2590F-066, R1C0;
OP 2201, R43, Attachments 6, 4, 2 and 3

NO Comments or Question Modification History at this time.

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

Number AA2.10 **RO** 3.7 **SRO** 3.7 **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 loss of cooling or seal injection

Question #: **78** **Question ID: 2018031** Rev. 0 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/17/2018 11:30:21 AM ☐ RO ☑ SRO **Origin: New** ☐ Past NRC Exam?

The plant has just completed a refueling outage. The reactor was brought critical and power was stabilized at 4%. The RCS is stable at 2250 psia and 532 °F. Then the "A" turbine bypass valve failed partially open. Reactor coolant temperature, as indicated by Tavg, lowers and stabilizes at 507 °F.

What effect will the lowering of temperature have on reactor power and what procedural required action will the Unit Supervisor direct.

☑ **A** Power lowers, restore Tavg to > 515 °F within 15 minutes, or place the plant in HOT STANDBY within the next 15 minutes

☐ **B** Power raises, restore Tavg to > 515 °F within 15 minutes, or place the plant in HOT STANDBY within the next 15 minutes

☐ **C** Power lowers, initiate emergency boration in accordance with AOP 2558, "Emergency Boration"

☐ **D** Power raises, initiate emergency boration in accordance with AOP 2558, "Emergency Boration"

Question Misc. Info: new question for 2018 NRC ILT exam, 2202, 2203

Justification

A - CORRECT: Power lowers and restore Tavg to > 515 °F within 15 minutes, or place the plant in HOT STANDBY within the next 15 minutes is correct. Coming out of a refueling outage at low power levels the Moderator Temperature Coefficient (MTC) will be slightly positive. This is due to having a high boron concentration and a very small level of xenon in the core. As temperature lowers the MTC gets more positive because the RCS coolant gets denser (RCS contracts). The effect of lowering RCS temperature with a positive MTC is to lower reactor power. Both the Plant Startup procedure, OP 2203 and the Technical Specification bases (under the minimum temperature for criticality) discuss positive MTC conditions at the beginning of core life. The Reactor Startup procedure OP 2202 and the Plant Startup procedure, OP 2203 provide direction for lowering RCS temperature. Specifically stating that if Tavg lowers to < 515 °F to restore temperature to > 515 °F within 15 minutes or place the plant in HOT STANDBY within the next 15 minutes. These procedures also state that if an uncontrolled cooldown occurred (Tc < 500 °F) then initiate emergency boration using AOP 2558.

B - INCORRECT: Power will not raise with a positive MTC when temperature lowers.
PLAUSIBLE: The candidate could think that power will rise because they are taught that power follows steam demand. This is normally the case. The steam demand will rise as the "A" turbine bypass valve opens further. But coming out of a refuel outage, on initial startup at very low power levels, MTC will be slightly positive.

C - INCORRECT: Power will lower but emergency boration is not required. Emergency boration is required when an uncontrolled cooldown occurs. An uncontrolled cooldown is defined as Tc less than 500 °F.
PLAUSIBLE: Procedural required actions contained in the Reactor Startup and Plant Startup procedures require emergency boration if an uncontrolled cooldown occurs where Tc lowers to less than 500 °F. The candidate could select this based on knowing it is a action they must take on a RCS uncontrolled cooldown.

D- INCORRECT: Power will not rise with a positive MTC if temperature lowers and emergency boration is not required. Emergency boration is required when an uncontrolled cooldown occurs. An uncontrolled cooldown is defined as Tc less than 500 °F.
PLAUSIBLE: The candidate could think that power will rise because they are taught that power follows steam demand. This is normally the case. The candidate could select to emergency borate based on knowing it is a action they must take on a RCS uncontrolled cooldown.

SRO justification: The first part of the question can be answered using RO knowledge of reactivity and startup procedures. The second part of the question can only be answered by an SRO applicant if he/she knows the information in Tech Spec bases (minimum temperature for criticality) and can assess the condition and select the appropriate procedural action (restore temperature or shutdown). No reference is provided. This question is linked to 10 CFR 55.43(b)(5).

References

Technical Specifications 3.1.1.5 , Minimum Temperature For Criticality and the associated bases
OP 2202, Reactor Startup ICCE
OP 2203, Plant Startup

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

Question #: **78** **Question ID:** **2018031** **Rev.** **0** ☐ **Student Handout?** ☑ **Lower Order?**

Last Edited: 6/17/2018 11:30:21 AM ☐ **RO** ☑ **SRO** **Origin:** **New** ☐ **Past NRC Exam?**

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

Number 2.1.43 **RO** 4.1 **SRO** 4.3 **CFR Link** (CFR: 41.10 / 43.6 / 45.6)

Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.

Question #: **79** **Question ID: 2018027** **Rev. 0** ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/19/2018 8:15:12 AM ☐ **RO** ☑ **SRO** **Origin: New** ☐ Past NRC Exam?

What are the conditions that will automatically actuate Auxiliary Feedwater and what is the Tech. Spec. Bases for those conditions?

☐ **A** When pressurizer pressure is greater than 2400 psia and Q-power is greater than 20% for greater than 10 seconds, to minimize catastrophic RCS pressure increase from ATWS and loss of feedwater

☑ **B** When steam generator narrow range level is less than 26.8% for greater than 3 minutes and 25 seconds, to avoid unacceptable return to power

☐ **C** When steam generator narrow range level is less than 26.8% for greater than 3 minutes and 25 seconds, to ensure Safety Limits are not violated.

☐ **D** When Q-power is greater than 20% for greater than 10 seconds and steam generator narrow range level is less than 26.8%, to avoid unacceptable return to power

Question Misc. Info: AAFW, Bases, Setpoints, time delay, 2537

Justification

A - INCORRECT: Q power power makes this answer incorrect. Power is sensed from control channel NI because this reactor trip must be independent of the RPS. This is another automatic actuation of AFW on high pressurizer pressure through the Diverse Scram System (DSS) and the ATWS Mitigating System Actuation Circuit (AMSAC). This automatic start of the AFW system is in response to indications of an Anticipated Transient Without a Scram (ATWS) event. The DSS is a redundant automatic reactor trip on high pressurizer pressure that is electrically independent of the Reactor Protection System (RPS). In addition to tripping the plant the DSS is sensed by the AMSAC which will provide for automatic actuation of AFW starting both electric pumps and opening both AFW regulating valves fully. If the plant is tripped on high pressurizer pressure (2400 PSIA as sensed by DSS) combined with a reactor power of greater than 20% (as sensed by control channel NIs), an automatic actuation of AFW will occur after a 10 second time delay. This actuation is designed to minimize the catastrophic RCS pressure increase resulting from an ATWS combined with a total loss of feedwater flow.
PLAUSIBLE: Auxiliary Feedwater will automatically actuate when pressurizer pressure is greater than 2400 psia and reactor power on control channel nuclear instruments is greater than 20% for 10 seconds to minimize catastrophic RCS pressure increase from ATWS and loss of feedwater.

B - CORRECT: Automatic initiation should occur only when feed is needed to recover level sufficiently and to avoid an unacceptable return to power. The three minute and 25 second delay time is sufficiently long so that, even if the affected steam generator is fed following a Design Basis Accident (DBA) Main Steam Line Break (MSLB), an unacceptable return to power situation does not occur. This delay allows the delayed neutron population to decrease sufficiently so as to avoid an unacceptable return to power during a casualty.

C - INCORRECT: Auxiliary feedwater does initiate when narrow range level is less than 26.8% fo greater than 3 minutes and 25 seconds but it is not to ensure Safety Limits are not violated.
PLAUSIBLE: It is reasonable for the examinee to surmise that the initiation of auxiliary feedwater would safety limits.

D - INCORRECT: Only steam generator narrow range level less than 26.8% for greater than 3 minutes and 25 seconds is used to automatically start auxiliary feedwater. It is the correct reason with an incorrect actuation logic. Q power is not used in any automatic auxiliary feedwater starting logic nor is any power level used with the 3 minute and 25 second time delay.
PLAUSIBLE: Steam generator narrow range level is less than 26.8% to avoid unacceptable return to power is right but must also include the 3 minutes and 25 second time delay to be correct. Reactor power greater than 20% by control channel nuclear instruments is used in the actuation logic for starting auxiliary feedwater to minimize catastrophic RCS pressure increase from ATWS and loss of feedwater. It is reasonable that an actuation logic to avoid an unacceptable return to power would have a power input.

SRO justification: The first part of the question can be answered using RO knowledge of auxiliary feedwater actuation criteria. The second part of the question can only be answered by an SRO applicant if he/she knows the information in Tech Spec bases (minimize catastrophic RCS pressure increase from ATWS and avoid unacceptable return to power). No reference is provided. This question is linked to 10 CFR 55.43(b)(2).

References

AFW-00-C Lesson Text, Auxiliary Feedwater System
AFW-00-C Lesson Text, Auxiliary Feedwater System, Figure 2, AFAS & AMSAC Interface

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 054 000054 (APE 54; CE E06) Loss of Main Feedwater /4

Number AA2.03 **RO** 4.1 **SRO** 4.2 **CFR Link** (CFR: 43.5 / 45.13)

AA2.03 Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Conditions and reasons for

Question #: **79**	**Question ID: 2018027**	**Rev. 0**	☐ **Student Handout?**	☑ **Lower Order?**	
Last Edited: 6/19/2018 8:15:12 AM	☐ **RO**	☑ **SRO**	**Origin: New**	☐ **Past NRC Exam?**	

AFW pump startup

Question #: **80** ***Question ID*: 1000053** **Rev.** 1 ☐ **Student Handout?** ☐ **Lower Order?**

Last Edited: 6/19/2018 8:50:00 AM ☐ ***RO*** ☑ ***SRO*** ***Origin*: Bank** ☐ **Past NRC Exam?**

Which one of the following describes the purpose or bases for the ESAS cabinet Backup Power Supplies?

☐ **A** Ensures the simultaneous loss of two Vital 120 VAC busses on the same Facility will NOT prevent an ESAS actuation if the loss occurred in concert with a Loss-Of-Offsite-Power and a design base accident.

☑ **B** Ensures the simultaneous loss of two Vital 120 VAC busses on the same Facility will NOT cause a premature SRAS, which would make the RWST unavailable if the loss occurred in concert with a Loss of Coolant Accident.

☐ **C** Ensures the simultaneous loss of a Vital 120 VAC bus and one Vital 4160 VAC bus on the same Facility will NOT prevent an ESAS actuation for any subsequent design base accident.

☐ **D** Ensures the simultaneous loss of a Vital 120 VAC source and a Vital DC bus on the same Facility will NOT cause a premature Auxiliary Feedwater actuation during an Excess Steam Demand event.

Question Misc. Info: MP2 LOUT ESA-01-C, ADM-02-J MB-2469, CFR55.41.7

Justification

A - INCORRECT: This statement is correct but it is not why we have the ESAS cabinet backup power supplies. A loss of two Vital 120 VAC busses will not prevent actuation of ESAS. One Vital 120 VAC busses supplies actuation power (VA-10) to facility 1 ESAS actuation cabinet 5 and the other Vital AC bus supplies actuation power (VA-20) to facility 2 ESAS actuation cabinet 6. Loss of one Vital AC on one Facillity will not affect actuation of ESAS on the opposite facility. The loss of offsite power will not take away power to safety equipment because an Emergency Diesel Generator will start and automatically power one vital 4160 VAC bus even with a loss of Facility of Vital AC.
PLAUSIBLE: The loss of one Vital 120 VAC Facility (two vital 120 VAC buses) will NOT prevent an ESAS actuation if the loss occurred in concert with a Loss-Of-Offsite-Power is plausible because it is true. One ESAS actuation cabinet would operate and start one complete train of safety equipment. But this would occur even without the ESAS sensor cabinet Backup Power Supplies.

B - CORRECT: Tech Spec bases state that the auctioneering circuit of the ESFAS sensor cabinets ensures that two sensor cabinets do not de-energize upon loss two Vital 120 VAC busses on the same Facility (loss of a DC bus), thereby resulting in the false generation of an SRAS. Power source VA-10 provides normal power to sensor cabinet A and backup power to sensor cabinet D. VA-40 provides normal power to sensor cabinet D and backup power to cabinet A. Power sources VA-20 and VA-30 and sensor cabinets B and C are similarly arranged. If the normal or backup power source for an ESFAS Sensor Cabinet is lost, two sensor cabinets would be supplied from the same power source, but would still be operating with no subsequent trip signals present. However, any additional failure associated with this power source would result in the loss of the two sensor cabinets, consequently generating a false SRAS. SRAS makes the RWST unavailable.

C - INCORRECT: This statement is correct but it is not why we have the ESAS cabinet backup power supplies. The design bases is that one complete train of power and equipment will be available. This is the case even without the ESAS sensor cabinet Backup Power Supplies.
PLAUSIBLE: The simultaneous loss of a Vital 120 VAC bus and one Vital 4160 VAC bus on the same Facility will NOT prevent an ESAS actuation for any subsequent design base accident is plausible because it is true. One ESAS actuation cabinet would operate and start one complete train of safety equipment.

D - INCORRECT: It is correct that a loss of a Vital 120 VAC source and a Vital DC bus on the same Facility will NOT cause a premature Auxiliary Feedwater actuation but this is not why we have ESAS Backup Power supplies.
PLAUSIBLE: It is correct that the simultaneous loss of a Vital 120 VAC source and a Vital DC bus on the same Facility will NOT result in a premature Auxiliary Feedwater actuation during an Excess Steam Demand.

SRO justification: This question meets the SRO level due to system knowledge (backup power supplies to sensor cabinets), design bases knowledge (need one complete train of vital 120 VAC, 125 VDC, 480 VAC, 4160 VAC, and equipment in a design base accident) and Tech Spec bases knowledge (ESAS backup power supplies are required to prevent a false SRAS on the loss of a single DC bus). No reference is provided. This question is linked to 10 CFR 55.43(b)(2).

References

Millstone Unit 2 Technical Specification, LCO 3.3.2.2 and associated bases.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 057 000057 (APE 57) Loss of Vital AC Instrument Bus / 6

Generic K/A Selected

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

Number 2.1.28 **RO** 4.1 **SRO** 4.1 **CFR Link** (CFR: 41.7)

G2.1.28 Knowledge of the purpose and function of major system components and controls.

Question #: **81** **Question ID:** **86243** Rev. 3 ☑ Student Handout? ☐ Lower Order?

Last Edited: 6/14/2018 8:35:20 AM ☐ RO ☑ SRO **Origin:** **Bank** ☐ Past NRC Exam?

Both Millstone Units are at 100% power.
The Turbine Driven Auxiliary Feedwater (TDAFW) Pump is out of service for maintenance and the applicable TSAS has been entered.

Then, ISO-New England/CONVEX declares a Capacity Deficiency Alert, and predicts that switchyard voltage would drop to 330 kV if either Millstone unit was lost.
C OP 200.8, "Response to ISO/CONVEX Alerts", has been entered and appropriate actions are being taken.

Which of the following describes the change in Technical Specification Action Statement, based on the given conditions?

☐ **A** Tech Spec LCO 3.0.3 should be entered because both bus 24C and bus 24D are NOT OPERABLE.

☑ **B** Tech Spec LCO 3.8.1.1 action d. should be entered because 2 Offsite Circuits are NOT OPERABLE.

☐ **C** Tech Spec LCO 3.0.3 should be entered because 2 Offsite Circuits and the TDAFW pump are NOT OPERABLE

☐ **D** Tech Spec LCO 3.8.1.1 action d. And 3.0.5 should be entered because 2 Offsite Circuits and the TDAFW pump are NOT OPERABLE.

Question Misc. Info: MP2*LOIT MB-00338, A80-01-C, AOP Degraded Voltage
Enabling Objective MB-05534

Justification

A - WRONG; TS 3.8.2.1 gives action requirements for "less than the above compliment of A.C. Busses" being OPERABLE, which would cover more than one vital VAC bus being out.
PLAUSIBLE; Student may remember busses 24C and 24D are required to be OPERABLE and believe that the loss of both vital AC bus operability would clearly not meet the TSAS.

B - CORRECT; If RSST voltage decreases to less than 3,900 volts, per SP 2619G-001, "One Offsite Circuit Inoperable" the RSST is inoperable. Per C OP 200.8, if ISO New England predicts grid voltage will drop below 345 KVA on loss of either Millstone Unit, both offsite lines are considered inoperable. Therefore, TSAS 3.8.1.1, Action d should be entered.

C - WRONG; TS 3.8.1.1 has a requirement for additional action of the loss of the TDAFW pump only if an emergency diesel generator is also inoperable.
PLAUSIBLE; Student may remember the requirement for the TDAFW pump in the offsite power TSAS but not how it relates to the applicability of the EDG requirement.

D - WRONG; The loss of the TDAFW pump is only considered an additional vulnerability by the Offsite Power Tech Spec if one of the EDGs is not available.
PLAUSIBLE; Student may considering the additional plant vulnerability and believe because the TS 3.8.1.1 does not cover a loss of the TDAFW pump with the loss of two offsite power sources, this must require TS 3.0.3 be applied.

References **Provided**

Handout; TS 3.8.1.1, A.C. Sources, Modes 1-4 and 3.8.2.1, Onsite Power Distribution Systems, Modes 1-4.
C OP 200.8, Response to ISO/CONVEX Alerts. St. 4.4

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 077 000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6

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)

G2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Question #: **82** **Question ID:** **1000038** **Rev.** 3 ☑ Student Handout? ☐ Lower Order?

Last Edited: 6/14/2018 8:36:11 AM ☐ **RO** ☑ **SRO** **Origin:** **Mod** ☐ Past NRC Exam?

The plant is in a normal 100% power lineup, when a Steam Generator tube leak developed.
AOP 2569, "Steam Generator Tube Leak", was entered and the leak rate was initially quantified at 8 gpm.

Present plant conditions are as follows:

- AOP 2575, "Rapid Downpower", has just been entered.
- The applicable actions are being taken to start a plant shutdown.
- A second charging pump was started.
- Letdown is now at 70 gpm and stable.
- Pressurizer level is 65% and stable.

Which one of the following describes the notification and Tech. Spec. Requirements, based on the existing conditions?

☑ **A** Notify the NRC within 4 hours due to a Tech. Spec. required shutdown.
Shutdown to Mode 3 within six (6) hours due to the RCS Leakage Tech. Spec.

☐ **B** Notifications made based on classification of an Unusual Event, Delta One.
Shutdown to Mode 3 within six (6) hours due to the RCS Leakage Tech. Spec.

☐ **C** Notify the NRC within 4 hours due to a Tech. Spec. required shutdown.
Shutdown to Mode 3 within four (4) hours due to the RCS Leakage Tech. Spec.

☐ **D** Notifications made based on classification of an Unusual Event, Delta One.
Shutdown to Mode 3 within four (4) hours due to RCS Leakage Tech. Spec.

Question Misc. Info: MP2 LOUT, CVC-01-C, MB-2360, 2001 Audit, NRC-2014

Justification

A - CORRECT; The RCS Leak has risen to 14 gpm (Charging flow minus Letdown flow minus RCP Seal Leakoff flow, or **88 gpm - 70 gpm - 4 gpm = 14 gpm**). This clearly exceeds the requirements of TS 3.4.6.2 Action "b", which requires the plant be in Hot Standby within 6 hours.
RAC 14, Attachment 1, “Plant Operation/ Equipment/Technical Specification Events”, requires NRC notification within 4 hours for a shutdown required by Tech. Specs.

B - WRONG; Primary-to-Secondary leakage must be > 25 gpm to classify as a UE/D-1.
PLAUSIBLE; Examinee may confuse the RCS leak rate classification requirement of > 10 gpm with the Primary-to-Secondary leak rate requirement.

C - WRONG: TS 3.4.6.2 action B requires to be in hot standby in 6 hours.
PLAUSIBLE; TSAS 3.4.6.2, Action "a" requires the RCS leakage be brought within Tech. Spec. Limits within 4 hours. Examinee may recognize correct Charging/Letdown flow mismatch requires a plant shutdown due to Tech. Specs., but not the correct TSAS time requirement.

D - WRONG; Primary-to-Secondary leakage must be > 25 gpm to classify as a UE/D-1 and TS 3.4.6.2 Action "b", requires the plant be in Hot Standby within 6 hours, not 4 hours.
PLAUSIBLE; Examinee may consider the RCS leak rate (> 10 gpm) drives a special state of Connecticut classification (Echo) and the NRC would be notified via the ENS.

SRO Justification: This question is SRO only as it requires assessing plant conditions and determining how changing conditions affect notification requirements and applicable TS action requirements of > 1 hour. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigating strategy of a procedure. This question is linked to 10 CFR 55.43(b)(5).

References **Provided**

Handout FAP06-002, MP2 EAL Tables.
TSAS 3.4.6.2, RCS Leakage. Also RAC 14, Attachment 1, “Plant Operation/ Equipment/Technical Specification Events”, requires NRC notification within 4 hours for a shutdown required by Tech. Specs.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 037 000037 (APE 37) Steam Generator Tube Leak / 3

Question #: **82** **Question ID:** **1000038** **Rev.** 3 ☑ **Student Handout?** ☐ **Lower Order?**

Last Edited: 6/14/2018 8:36:11 AM ☐ **RO** ☑ **SRO** **Origin:** **Mod** ☐ **Past NRC Exam?**

Number AA2.04 **RO** 3.4 **SRO** 3.7 **CFR Link** (CFR: 43.5 / 45.13)

AA2.04 Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: Comparison of RCS fluid inputs and outputs, to detect leaks

Question #: N/A | Question ID: 1000038 | Rev. 1 | ☐ Student Handout? | ☐ Lower Order?
☐ RO ☑ SRO | Origin: Parent | ☑ Past NRC Exam?

PARENT

The plant is in a normal 100% power lineup, when a Steam Generator tube leak develops.

Present plant conditions are as follows:

- Letdown has lowered to 32 gpm and stabilized
- No backup charging pumps are running
- Pressurizer level is 64% and stable

The crew then takes the applicable actions to begin a plant shutdown.

Which ONE of the following conditions would indicate that the Steam Generator tube leak rate has risen to a new stable value and when is the plant required to be in HOT STANDBY per the TS?

☐ **A** After starting the second charging pump, the RO adjusts letdown flow until pressurizer level stabilizes at 65%. Letdown flow rises to 70 gpm and stabilizes.
Be In HOT STANDBY within 4 hours.

☐ **B** After starting the second charging pump, the RO adjusts letdown flow to 84 gpm. Pressurizer level then lowers to 63.5% and letdown flow lowers to 76 gpm before both stabilize.
Be In HOT STANDBY within 4 hours.

☑ **C** After starting the second charging pump, the RO adjusts letdown flow until pressurizer level stabilizes at 65%. Letdown flow rises to 70 gpm and stabilizes.
Be In HOT STANDBY within 6 hours.

☐ **D** After starting the second charging pump, the RO adjusts letdown flow to 84 gpm. Pressurizer level then lowers to 63.5% and letdown flow lowers to 76 gpm before both stabilize.
Be In HOT STANDBY within 6 hours.

Question Misc. Info: MP2 LOUT, CVC-01-C, MB-2360, 2001 Audit, NRC-2014

Justification

A - WRONG; TS 3.4.6.2 Action B requires to be in hot standby in 6 hours.
Plausible; Examinee may recognize correct Charging/Letdown flow mismatch, but not the correct TSAS time requirement.

B - WRONG; Initial leak rate from conditions given is 8 gpm based on Charging/Letdown mismatch with 4gpm RCP bleed off flow. Manually raising letdown flow to match the new charging flow will force the pressurizer level control system to readjust letdown flow to account for the 8 gpm leak. Leak rate is still 8 gpm in the answer choice. TS 3.4.6.2 action B requires to be in hot standby in 6 hours.
Plausible; Examinee may consider the starting of the only available backup charging pump with the indicated Charging/Letdown mismatch would require an immediate shutdown, but not know the TSAS time requirement.

C - CORRECT: PZR level stabilizes with the new charging flow (88gpm). Letdown flow stabilizes more than 8 gpm below the expected charging flow for two pumps. Based on Charging/Letdown mismatch, the leak rate has gone from 8 gpm to 14 gpm. TS 3.4.6.2 action B requires to be in hot standby in 6 hours.

D - WRONG; Initial leak rate from conditions given is 8 gpm based on Charging/Letdown mismatch with 4gpm RCP bleed off flow. Manually raising letdown flow to match the new charging flow will force the pressurizer level control system to readjust letdown flow to account for the 8 gpm leak. Leak rate is still 8 gpm in the answer choice.
Plausible; Examinee may consider the starting of the only available backup charging pump meets the TSAS requirements to go to Mode 3.

SRO Justification: This question is SRO only as it requires assessing plant conditions and determining how conditions are changing and applying TS knowledge of > 1 hour action requirements. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigating strategy of a procedure. This question is linked to 10 CFR 55.43(b)(5).

Question References not yet listed.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 037 Steam Generator (S/G) Tube Leak

Number AA2.12 **RO** 3.3 **SRO** 4.1 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: Flow rate of leak

Question #: ***N/A*** ***Question ID:*** ***1000038*** **Rev.** **1** ☐ **Student Handout?** ☐ **Lower Order?**

☐ ***RO*** ☑ ***SRO*** ***Origin:*** ***Parent*** ☑ **Past NRC Exam?**

Question #: **83** **Question ID: 2018041** **Rev. 0** ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/15/2018 2:37:42 PM ☐ **RO** ☑ **SRO** **Origin: New** ☐ Past NRC Exam?

A fuel pin has ruptured at power due to a recently disclosed manufacturing defect. TSAS 3.4.8, RCS Specific Activity LCO, has forced a plant cooldown to replace the leaking fuel bundle.
Preparations are presently being made to place the Shutdown Cooling (SDC) System in service, in accordance with OP 2207, Plant Cooldown. A containment purge is not required at this time.
Due to the existing RCS activity levels, Chemistry supervision recommends taking the additional action described in OP 2207 to limit the unintentional and unmonitored release of fission product gasses, prior to aligning the SDC system to the RCS.

Which of the following actions would the US direct, based on the Chemistry supervisors recommendation?

☐ **A** Ensure all Main Exhaust fans and Aux. Building fan F-17 are placed in service.

☐ **B** Direct the Enclosure Building Filtration system is placed in service, aligned to the Enclosure Building.

☐ **C** Direct the Enclosure Building Filtration system is placed in service, aligned to the Spent Fuel Pool area.

☑ **D** Direct the RWST Ventilation Filtration System be installed and placed in service.

Question Misc. Info: MP2*LOIT SDC, LOCA, Fuel

Justification

A - WRONG; Although the Main Exhaust fans (and F-17) take a suction on the Aux. Building, discharge to the Unit 2 stack and are monitored by the Kaman rad monitor, there is no guidance in OP 2207 to start additional fans if RCS activity is exceptionally high.
PLAUSIBLE; Student may assume that because the Main Exhaust fans (and F-17) take a suction on the Aux. Building rad. Waste areas, and would ventilate and monitor an accidental gaseous release from the waste gas storage tanks, that they would be key to preventing an unmonitored release in this scenario.

B - WRONG; OP 2207 does not direct EBFAS be placed in service, regardless of the RCS activity levels, because it's filters would quickly become exhausted unnecessarily. EBFAS would only be used if it was necessary to purge CTMT.
PLAUSIBLE; The student may realize that the act of forcing PZR sprays during the cooldown would increase CTMT activity, especially with the existing RCS activity levels. EBFAS would filter out any fission products that escape CTMT when it is eventually opened for fuel movement.

C - WRONG; OP 2207 does not direct EBFAS be placed in service, regardless of the RCS activity levels, because it's filters would quickly become exhausted unnecessarily. However, this may be done during fuel movement if recommended by HP due to high airborne activity.
PLAUSIBLE; Student may believe it is necessary to pull this step into OP 2207 due to the leaking fuel pin.

D - CORRECT; The SDC (LPSI) pumps recirc back to the RWST, therefore, any gaseous activity in the RCS will vent into the RWST. This would result in an unmonitored release of gaseous activity because the RWST vents directly to the environment. OP 2207 Plant Cooldown, states that the RWST Ventilation Filtration System should be installed and placed in service for planned/refuel outages prior to opening SI---651 and SI---652. The need to install the filter are driven by the fact that the RCS activity Tech. Spec is forcing the plant to Mode 5 due to a leaking fuel bundle.

SRO Justification: OP 2207 is a very complex procedure that intertwines several specific system procedures and unusual alignments during the process of going from Mode 3 to Mode 5. This question is linked to 10 CFR 55.43(b)(5).

References

OP 2207, R43C0, St. 3.7.8 Note

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 060 Accidental Gaseous Radwaste Release

Number AA2.06 **RO** 3.6* **SRO** 3.8 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Accidental Gaseous Radwaste: Valve lineup for release of radioactive gases

Question #: **84** **Question ID:** 1000202 Rev. 1 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/15/2018 3:48:52 PM ☐ RO ☑ SRO **Origin:** Bank ☐ Past NRC Exam?

A plant heatup is in progress with RCS temperature presently at 250 °F and slowly rising.
The Instrument Air supply to the Containment Air Receiver must be isolated to repair an air leak on the supply piping outside of containment.
In order to maintain air to containment for valve operation, 2- SA-19, "Containment Station Air Header Isolation", must be opened to align Station Air to the Containment Air Receiver.

Which of the following describes the administrative requirements of opening SA-19 at this time?

☐ **A** There are no restrictions on opening SA-19 at this time, however the containment Technical Specifications requires the valve be closed prior to RCS temperature exceeding 300°F.

☐ **B** When SA-19 is opened, all containment air operated safety related valves become inoperable, therefore it cannot remain open more than one hour with RCS temperature above 200°F.

☑ **C** Opening SA-19 should be minimized to prevent moisture build up in the Containment Instrument Air System, however RCS heatup to NOP/NOT is allowed provided a Dedicated Operator is stationed at the valve while it is opened.

☐ **D** Opening SA-19 should be minimized to prevent moisture build up in the Containment Instrument Air System, a Dedicated Operator is required at the valve while it is opened only if RCS temperature is going to be raised above 300°F.

Question Misc. Info: MP2*LOIT (SAS-01-C) 2332A, SA, IA, CTMT

Justification

A - WRONG; Based on TS 3.6.3.1, while in Mode 4 or above a Dedicated Operator is required to be stationed at the valve while it is opened.
PLAUSIBLE; Student may believe restrictions on this valve are similar to those mentioned in TS 3.6.3.1 for the MSIVs, due to the air system being a seldom used backup.

B - WRONG; Opening this valve does not inop all TSAS required valves in CTMT because TS 3.6.3.1 does not specify Instrument Air be the motive force to operate the valves and air operated CTMT isolation valves fail to their accident position on loss of air or control power.
PLAUSIBLE; Student may feel the potential contamination of CTMT valves due to being supplied by an air system other than what was designed, would render them inoperable and put the plant into TSAS 3.0.3.

C - CORRECT; IAW OP 2332A, Station Air, opening SA-19 should be minimized due to the potential contamination of air operated valves in CTMT. Also, CTMT Integrity Tech. Specs. 3.6.3.1 Bases states that as long as these administrative requirements of a dedicated operator being stationed at the open valve are met, it is NOT necessary to log into any TSAS. Therefore, the RCS heatup to Mode 3 (>300°F) may continue.

D - WRONG; While it is true that the RCS heatup to Mode 3 (>300°F) may continue with a dedicated operator stationed at the valve, opening the valve without the dedicated operator in Mode 4 puts the plant in a 4 hour TSAS. Procedure requirements state a dedicated operator must be stationed at the valve whenever it is opened in Mode 4 or above (>200°F).
PLAUSIBLE; Student may remember the use of a dedicated operator meets the requirements of TS 3.6.3.1 for CTMT Isolation valves but feel it is not immediately necessary when the valve is opened because the TSAS has a 4 hour time limit.

SRO Justification: This question is SRO only as it requires assessing plant conditions and determining the administrative requirements of the Containment Integrity Tech Spec, as stated in its Bases. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigating strategy of a procedure. This question is linked to 10 CFR 55.43(b)(2).

References

OP 2332A, Station Air, Caution for St. 4.4.1 and TS 3.6.3.1, CTMT Integrity

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 069 000069 (APE 69) Loss of Containment Integrity / 5

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)

G2.1.32 Ability to explain and apply system limits and precautions.

Question #: **85** ***Question ID:*** ***2018004*** **Rev.** **0** ☐ **Student Handout?** ☐ **Lower Order?**

Last Edited: 6/20/2018 10:12:20 PM ☐ ***RO*** ☑ ***SRO*** ***Origin:*** ***New*** ☐ **Past NRC Exam?**

The unit has experienced a Excess Steam Demand event. The crew completed Standard Post Trip Actions, EOP 2525 and is currently in the Excess Steam Demand Event procedure, EOP 2536.

Then a fire in occurs. The fire is not affecting equipment required for Safe Shutdown. Assistance from Unit 3 has been requested.

How will the Unit Supervisor (US) direct the Fire procedure AOP 2559 with the Excess Steam Demand Event procedure EOP 2536?

☐ **A** AOP implementation in parallel with the EOP. With the AOP as the primary controlling procedure.

☐ **B** Exit the EOP and enter the AOP. Procedures prohibit being in both an AOP and EOP.

☑ **C** AOP implementation in parallel with the EOP. With the EOP as the primary controlling procedure.

☐ **D** Unit 3 will enter their Fire AOP and fight the fire.

Question Misc. Info: MP2*ILT AOP 2504, VA-10, 2260

Justification

A - INCORRECT: AOP implementation in parallel with the EOP. With the AOP as the primary controlling procedure. This is incorrect. The AOP is not the primary controlling procedure while the EOP is still in affect.
PLAUSIBLE: Since AOPs and an EOP are implemented in parallel. And the AOP does become the controlling procedure once the EOP exit criteria are met.

B - INCORRECT: Exit the EOP and enter the AOP. Procedures prohibit being in both an AOP and EOP. This is incorrect. The EOP will not be exited. The EOP is the primary controlling procedure.
PLAUSIBLE: The examinee could reason that an AOP condition might be a more immediate and larger challenge to the plant and that since you can't be in two or more Optimal Recovery EOPs at the same time that you can't be in an EOP and AOP at the same time.

C - CORRECT: AOP implementation in parallel with the EOP. With the EOP as the primary controlling procedure. The AOP users guide directs parallel performance of designated AOP actions at the appropriate point in EOP implementation. It further states that if the AOP remains appropriate, the EOP is the primary controlling procedure until the EOP exit criteria are met.

OP 2255 step 1.11 is the Parallel Performance of EOP and AOP Actions section. It states that the Unit Supervisor directs parallel performance of designated AOP actions at the appropriate point in EOP implementation. It further states that if the AOP remains appropriate, the EOP is the primary controlling procedure until the EOP exit criteria are met. And if the EOP exit criteria are met and AOP actions remain appropriate, the AOP becomes the primary controlling procedure.

D - INCORRECT: Unit 3 will enter their Fire AOP and fight the fire. This is incorrect. The AOP will be entered with the EOP. Under some concurrent events certain additional AOP actions may be needed. The EOPs do not contain guide for a fire. Unit 3 may be directed to fight the fire as the lead but the unit 2 Fire AOP would still be entered.
PLAUSIBLE: It is plausible that Unit 3 would take the lead on the fire. And if they had the lead that they would use their procedure and Unit 2 would not enter their AOP.

SRO justification: This question meets the SRO level. The SRO examinee must assess the facility condition and select appropriate procedures during an emergency situation. No reference is provided. This question is linked to 10 CFR 55.43(b)(5).

References

OP 2255 Unit 2 AOP Upgrade User's Guide

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.4.8 **RO** 3.8 **SRO** 4.5 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

G2.4.8 Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Question #: **86** **Question ID:** **84895** **Rev.** **2** ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/14/2018 8:38:36 AM ☐ **RO** ☑ **SRO** **Origin:** **Bank** ☐ Past NRC Exam?

Shutdown Cooling flow has just been lost and a standby pump is available and aligned.

Which of the following conditions would allow the US to direct the immediate start of the standby LPSI pump, once all applicable system valves are appropriately aligned?

☐ **A** The SDC flow loss is known to be due to RCS level dropping below the centerline.

☐ **B** The SDC flow loss is causing an RCS pressure rise that will prevent gravity feed makeup.

☑ **C** The SDC flow loss is known to be due to the running LPSI pump tripping on overload.

☐ **D** The SDC flow loss is causing RCS temperature to rise > 10 °F and exceed 200 °F.

Question Misc. Info: MP2*LORT SDC, 2572

Justification

A - WRONG; A Caution in the beginning of Section 4.0 gives guidance NOT to start the standby pump before recovering NPSH, to avoid damage to the remaining pump.
PLAUSIBLE; Student may remember there is an exception to the hold on starting the standby pump that pertains to the RCS about to boil. However, even this exemption requires verifying adequate LPSI suction pressure.

B - WRONG; This is NOT a reason described in the AOP to chance losing the standby pump.
PLAUSIBLE; Student may remember the AOP treats this decision point as a critical one in that action must be taken in a timely manner to prevent this option from being lost.

C - CORRECT; Section 5.0 is utilized when the loss of SDC was known to be caused by the tripping of the running pump. It gives direction to start the available standby pump as soon as the system is aligned.

D - WRONG; Although this would result in an Alert classification, it is NOT a reason to chance losing the standby pump.
PLAUSIBLE; Student may recognize the classification level criteria and feel that a plant condition that calls for an Alert classification (entire SERO is activated) would be grounds enough to pull the step forward to restore SDC flow.

SRO Justification: This question is SRO only as it requires assessing various detailed mitigating strategies and making a judgement call as to the specific one to use in mitigating the event. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigating strategy of a procedure. This question is linked to 10 CFR 55.43(b)(5).

References

AOP 2572, Loss of Shutdown Cooling, R14C0, Sect. 5

NO Comments or Question Modification History at this time.

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

Number A2.01 **RO** 2.7 **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: Failure modes for pressure, flow, pump motor amps, motor temperature, and tank level instrumentation

Question #: **87** **Question ID: 2018033** **Rev. 0** ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/20/2018 10:13:11 PM ☐ **RO** ☑ **SRO** **Origin: New** ☐ Past NRC Exam?

The plant is in Mode 6 with Refuel Pool level 16 feet above the Reactor Vessel flange. Which of the following equipment, if removed from service, WOULD violate Technical Specification requirements?

- ☐ **A** One Emergency Diesel Generator
- ☑ **B** One RBCCW header
- ☐ **C** One Service Water header
- ☐ **D** The Reserve Station Service Transformer

Question Misc. Info: MP2*ILT RBCCW, SDC, Bases

Justification

Technical Specification (3.9.8.2) requires two shutdown cooling trains be OPERABLE and one shutdown cooling train be in operation when in MODE 6 with water level < 23 feet above the top of the reactor vessel flange. A note in the Technical Specification LCO states that "The normal or emergency power source may be inoperable for each shutdown cooling train." The TS bases requires two OPERABLE SDC trains, and is delineated as; 2 SDC pumps, 2 SDC heat exchangers, 2 RBCCW pumps, 2 RBCCW heat exchangers, and 2 SW pumps in MODE 6. In addition, 2 RBCCW headers are required to provide cooling to the SDC heat exchangers, but only 1 SW header is required to support the SDC trains. Technical Specification (3.9.8.1) require one shutdown cooling train be OPERABLE and in operation when in MODE 6 with water level $\geq$ 23 feet above the top of the reactor vessel flange.

A - INCORRECT: One Emergency Diesel Generator is not correct. For a SDC training to be OPERABLE it must have OPERABLE either it's emergency or normal power source. Technical Specification (3.9.8.2) requires two shutdown cooling trains be OPERABLE and one shutdown cooling train be in operation when in MODE 6 with water level < 23 feet above the top of the reactor vessel flange. A note in the Technical Specification LCO states that "The normal or emergency power source may be inoperable for each shutdown cooling train."
PLAUSIBLE: The examinee should understand that with the refuel pool less than 23 feet that two Shutdown cooling trains are required to be Operable. And that for a Shutdown cooling train to be Operable it must have it's emergency power source.

B - CORRECT: One RBCCW header is correct. The TS bases requires two OPERABLE SDC trains, and is delineated as; 2 SDC pumps, 2 SDC heat exchangers, 2 RBCCW pumps, 2 RBCCW heat exchangers, and 2 SW pumps in MODE 6. In addition, 2 RBCCW headers are required to provide cooling to the SDC heat exchangers.

C - INCORRECT: One Service Water header is not correct. The TS bases requires two OPERABLE SDC trains, and is delineated as; 2 SDC pumps, 2 SDC heat exchangers, 2 RBCCW pumps, 2 RBCCW heat exchangers, and 2 SW pumps in MODE 6. In addition, 2 RBCCW headers are required to provide cooling to the SDC heat exchangers, but only 1 SW header is required to support the SDC trains.
PLAUSIBLE: The examinee should understand that with the refuel pool less than 23 feet that two Shutdown cooling trains are required to be Operable. And that for a Shutdown cooling train to be Operable it must have an independent Service Water train supporting it.

D- INCORRECT: The Reserve Station Service Transformer is not correct. Technical Specification (3.9.8.2) requires two shutdown cooling trains be OPERABLE and one shutdown cooling train be in operation when in MODE 6 with water level < 23 feet above the top of the reactor vessel flange. A note in the Technical Specification LCO states that "The normal or emergency power source may be inoperable for each shutdown cooling train."
PLAUSIBLE: The examinee should understand that with the refuel pool less than 23 feet that two Shutdown cooling trains are required to be Operable. And that for a Shutdown cooling train to be Operable it must have it's normal offsite power source.

SRO justification: The question can only be answered by an SRO applicant if he/she knows the information in Tech Specs (SDC loop operability requirements with and without refuel pool level $\geq$ 23 feet above the reactor vessel flange) and the Tech Spec bases associated (only one service water header is required to support 2 operable SDC trains, an operable SDC train only requires and emergency or normal power source, and only one 125 VDC bus is required to support 2 SDC trains). No reference is provided. This question is linked to 10 CFR 55.43(b)(2).

References

Technical Specifications 3.9.8.1 and 3.9.8.2
Technical Specification Bases for 3/4.9.8
Technical Specifications 3.8.2.2
Technical Specifications 3.8.2.3
Technical Specifications 3.7.4.1

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 008 008 (SF8 CCW) Component Cooling Water

Question #: **87** **Question ID: 2018033** **Rev. 0** ☐ **Student Handout?** ☐ **Lower Order?**

Last Edited: 6/20/2018 10:13:11 PM ☐ **RO** ☑ **SRO** **Origin: New** ☐ **Past NRC Exam?**

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)

G2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Question #: **88** **Question ID: 2018034** Rev. 1 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/15/2018 2:48:09 PM ☐ RO ☑ SRO **Origin: Mod** ☐ Past NRC Exam?

Which one of the following describes a scenario and reason, where an EOP mitigating strategy taken to preserve or restore a Safety Function, would violate an applicable plant Tech. Spec.?

☐ **A** If a CEA did not insert, Shutdown Margin would be violated if pressurizer level is rising above the existing setpoint, in order to preserve RCS Pressure Control.

☐ **B** If a loss of normal and auxiliary feedwater flow were to occur, RCS cooldown rate would be violated in order to establish feedwater flow with a condensate pump.

☑ **C** If all feedwater flow was lost and could not be recovered, pressurizer maximum level requirements would be violated in order to restore RCS Heat Removal.

☐ **D** If all Vital AC power was lost and power was available from Unit 3, electrical facility separation requirements would be violated in order to restore Vital Auxiliaries.

Question Misc. Info: MP2*ILT 2260, SFSC

Justification

A - WRONG; With multiple CEAs not inserted, numerous procedures require "Emergency Boration" be initiated until Shutdown Margin is verified by calculation and Chemistry verification of the RCS boron concentration. However, the EOP Users Guide, OP 2260, allows for early termination of boric acid injection if "Pressurizer level is approaching the upper end of the control band", which is 35% - 70% in the EOPs. The shutdown PZR level setpoint is 40%.
PLAUSIBLE; Student may recall that OP 2260, EOP Users Guide, allows the required boric acid injection flow to be suspended, if rising PZR level is approaching the upper limit of 70% and the danger of PTS exists if PZR pressure control is lost.

B - WRONG; The RCS must only be cooled down to 515°F in order to isolate the SGTR and restore CTMT Isolation. The Cooldown Limit is < 100°F/hour, which would not be reached (or violated) if the RCS cooldown stopped at 515°F for SGTR isolation, as required by the EOP.
PLAUSIBLE; Student may confuse the affect of the RCS cooldown due to the ESD as compounding the impact of the required RCS cooldown for SGTR isolation.

C - CORRECT; If all feedwater flow was lost and unrecoverable, the only option left is Once-Through-Cooling, which involves opening the PORVs and overriding off all PZR heaters. The PZR would then fill solid, well above the Tech. Spec. Limit of 70% and both Tech. Spec. required banks of PZR heaters are made inoperable by opening their breakers.

D - WRONG; Recovering Vital AC power by cross-ting busses with Unit 3 does not violate the Tech. Spec. electrical facility separation requirement for AC electrical busses because the Unit 3 cross-tie is part of the Offsite Transmission Network and is not part of the facility separation requirements.
PLAUSIBLE; Student may consider cross-tying a Vital AC bus with another Unit to violate the Tech. Spec. Required separation of electrical systems.

SRO Justification: This question is SRO only as it requires assessing various mitigating strategies and making a judgement call as to how they interact with the Operability requirements of various Tech. Specs. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigating strategy of a procedure. This question is linked to 10 CFR 55.43(b)(2) and 10 CFR 55.43(b)(5).

References

EOP 2537, R25C0, St. 6, C.A. For OTC;
TS 3.4.4, Pressurizer.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 010 010 (SF3 PZR PCS) Pressurizer Pressure Control

Generic K/A Selected

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

Number 2.4.22 **RO** 3.6 **SRO** 4.4 **CFR Link** (CFR: 41.7 / 41.10 / 43.5 / 45.12)

G2.4.22 Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

Question #: **N/A** **Question ID: 2014052** Rev. 0 ☐ Student Handout? ☐ Lower Order?

☐ RO ☑ SRO **Origin: Parent** ☑ Past NRC Exam?

PARENT

While in EOPs, maintaining the Safety Function is of a higher priority than maintaining Technical Specification (TS) compliance for the following reason:

☐ **A** Technical Specifications are limits intended to ensure that plant configuration at the start of an accident is consistent with Design Basis Accident assumptions. Safety functions ensure that acceptable fuel design limits are not exceeded during implementation of EOPs.

☐ **B** Technical Specifications are limits only during plant operation and are not applicable during an accident. Safety functions ensure that acceptable fuel design limits are not exceeded during implementation of EOPs.

☐ **C** Technical Specifications are limits only during plant operation and are not applicable during an accident. Safety functions prevent core damage or minimize radiation releases to the general public during an accident.

☑ **D** Technical Specifications are limits intended to ensure that plant configuration at the start of an accident is consistent with Design Basis Accident assumptions. Safety functions prevent core damage or minimize radiation releases to the general public.

Question Misc. Info: MP2*LOIT, NRC-2014

Justification

A - Incorrect: GDC 20, Protection System Functions are designed to ensure acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences. The purpose of Safety Functions is to provide a set of conditions or actions needed to prevent core damage or minimize radiation releases to the general public. Depending on the accident sequence and whether or not it is beyond design basis will determine whether or not fuel limits are exceeded. The safety functions do not by themselves ensure fuel limits are maintained.
Plausible: The applicant may know the purpose of Technical Specifications but not understand the purpose of Safety Functions.

B - Incorrect: Technical Specifications do not only apply during plant operation. During an emergency, the crew's attention has to be directed to completing the procedure to mitigate the event, and not be distracted by documenting T/S LCO Action Statements. Documentation of TSAS entered can be reconstructed later from procedures and rough logs.
Plausible: Applicant may believe that Technical Specifications do not apply during an accident.

C - Incorrect: Technical Specifications do not only apply during plant operation. During an emergency, the crew's attention has to be directed to completing the procedure to mitigate the event, and not be distracted by documenting T/S LCO Action Statements. Documentation of TSAS entered can be reconstructed later from procedures and rough logs.
Plausible: Applicant may believe that Technical Specifications do not apply during an accident.

D - CORRECT: This is the basis for prioritizing Safety Functions over Technical Specifications during EOP usage from the definition of Safety Function and Section 1.5 Technical Specifications in OP 2260 EOP User Guide.

SRO Only Justification: This question is SRO only as it requires knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures. The question cannot be answered by solely knowing systems knowledge, Technical Specification Safety Limits or LCO information listed in the LCO statement.

Question References not yet listed.

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.4.22 **RO** 3.6 **SRO** 4.4 **CFR Link** (CFR: 41.7 / 41.10 / 43.5 / 45.12)

G2.4.22 Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

Question #: **89** **Question ID: 2018035** **Rev.** 0 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/8/2018 3:22:55 PM ☐ **RO** ☑ **SRO** **Origin: New** ☐ Past NRC Exam?

The plant is at 100% power, steady state, when an inadvertent SIAS, CIAS and EBFAS occurs on Facility 1.
The required actions of AOP 2585, "Immediate Operator Actions", have been completed.
AOP 2571, "Inadvertent ESFAS Actuation" has just been entered.

Which of the following conditions and mitigating actions would be a priority for the US to address?

☑ **A** The BOP reports a Stator Water Cooling high temperature alarm.
Restore the Service Water System flow path to normal.

☐ **B** The RO reports 4 CAR Fans are running, 2 in fast and 2 in slow.
Restore CAR Fan operation to normal.

☐ **C** The RO reports high seal pressure alarms on all four RCPs.
Restore RCP Seal Bleedoff flow path to normal.

☐ **D** BOP reports the Condenser Air Removal flow path is isolated.
Restore Condenser Air Removal System flow path to normal.

Question Misc. Info: MP2*ILT AOP 2571, ESAS, SIAS

Justification

A - CORRECT; SIAS actuation will isolate Service Water (SW) flow to the TBCCW heat exchangers and divert SW flow to the RBCCW heat exchangers. TBCCW is the heat sink for the Stator Water Cooling System, which will trip the Main Generator on a high temperature. That is why realignment of SW flow is the next action to be taken after attempting to reset the inadvertent action signal.

B - WRONG; The CTMT ventilation system is not designed to handle four CAR fans in operation if all four are operating in "fast" speed. Normal operation requires three fans running in fast speed (the fourth fan is secured) in order to maintain CTMT temperature. Although the inadvertent ESFAS actuation will have possibly started the fourth fan, it would also have shifted two of them to "slow" speed.
PLAUSIBLE; Student may have misunderstood the limitations on the CTMT ventilation system design criteria and therefore recognized that as a priority to prevent the loss of a Tech. Spec. system.

C - WRONG; The seal pressure alarms are a result of RCP bleedoff flow being diverted to the PDT by way of a relief valve. The system is designed to handle this without causing a failure of the RCP seals.
PLAUSIBLE; Student may consider seal pressure alarms on all four RCPs as reason to quickly rectify the cause before a catastrophic seal failure occurs.

D - WRONG; This condition will eventually cause a loss of condenser vacuum, but it will take a long time. The Stator Water Cooling issue is the overriding priority.
PLAUSIBLE; Student may recognize this as something that must be corrected and is the first item to be addressed by AOP 2571 under the Inadvertent EBFAS Actuation section.

SRO Justification: This question is SRO only as it requires knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures. The question cannot be answered by solely knowing systems knowledge, Technical Specification Safety Limits or LCO information listed in the LCO statement. This question is linked to 10 CFR 55.43(b)(5)

References

AOP 2585, R3C0, IOA, St. 16;
AOP 2571, Inadvertent ESFAS Actuation.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 013 013 (SF2 ESFAS) Engineered Safety Features Actuation

Number A2.06 **RO** 3.7* **SRO** 4.0 **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

A2.06 Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertent ESFAS actuation

Question #: **90** **Question ID: 2018036** Rev. 0 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/25/2018 1:35:19 PM ☐ RO ☑ SRO **Origin: New** ☐ Past NRC Exam?

A plant trip occurred as a result of a Large Break Loss of Coolant Accident (LBLOCA). The following is the current plant status:

- EOP 2525, Standard Post Trip Actions is being implemented by the crew
- Bus 24C failed to transfer to the RSST on the trip
- Containment pressure is 31 psig and rising
- Vital Instrument bus VA-10 was lost on the trip and Facility 1 components did not actuate
- SIAS, CIAS, EBFAS, MSI and CSAS alarms are in on C01
- All Facility 2 emergency equipment responded as designed

What Containment Cooling equipment is available and what procedural actions will the Unit Supervisor take to ensure that ALL available Containment Cooling equipment is operating?

☐ **A** US will ensure Facility 2 CS and CAR fans are operating in EOP 2525, then in EOP 2532 the US will direct restoration of power to 24C and the start of "A" CS pump and "A" & "C" CAR fans.

☑ **B** US will ensure the RO places the facility 1 RBCCW in PTL, the "A" EDG is powering 24C, the "A" Service Water pump is operating, and ensure that both CS pumps and "B" & "D" CAR fans have started in EOP 2525

☐ **C** US will ensure the RO manually starts the "A" EDG to power 24C, the "A" Service Water pump is operating, and ensure that both CS pumps and all four CAR fans have started in EOP 2525.

☐ **D** US will ensure Facility 2 CS and CAR fans are operating in EOP 2525, then in EOP 2532 the US will direct restoration of power to 24C and the start of "A" CS pump.

Question Misc. Info: MP2*ILT, 2525, 2532, VA-10, service water

Justification

A - INCORRECT: Both CS pumps are available. But all four CAR fans are not available. The "A" & "C" CAR fans will not be available because they will not have RBCCW to them. EOP 2525, SPTAs places the RBCCW pump on the facility without power to PTL when containment pressure is ≥ 20 psig. Therefore the facility 1 RBCCW pump will be placed in PTL and the "A" & "C" CAR fans will have no cooling to them, making them unavailable. The restoration of power to 24C will also occur in EOP 2525 not EOP 2532. EOP 2525 directs if either bus 24C or 24D is not energized then attempt to restore power by starting the EDG and closing it's output breaker. Once 24C is energized the procedure directs ensuring that CSAS has actuated. And ENSURE means to start the "A" CS pump. The "A" CS pump will not be running once power is restored to 24C because VA-10 is loss. VA-10 is actuation power for facility 1 ESF equipment. With power to 24C and no power on VA-10 the "A" CS pump will be started by closing the breaker from the handswitch on C01.
PLAUSIBLE: EOP 2525 is not written assuming the loss of VA-10. AOPs are generally not used concurrently with EOP 2525 SPTAs. AOPs are used with EOPs entered after the SPTAs as long as they don't hinder the performance of the EOP. Also design bases requires only one complete train in a design bases accident, therefore having only one train in EOP 2525 is sufficient.

B - CORRECT: Both CS pumps and "B" & "D" CARs are available and will be running in EOP 2525. EOP 2525 SPTA will ensure the RO places the facility 1 RBCCW in PTL, manually starts the "A" EDG to power 24C, ensures the "A" Service Water pump is operating, and ensures both CS pumps and "B" & "D" CAR fans have started in EOP 2525. Both 4160 VAC are checked. If they are not energized the RO will manually start and close the breaker to power the bus that was not energized per procedure EOP 2525 response not obtained actions. It is the procedural direction to attempt to energize both 24C & 24D if either or both are not energized.

C - INCORRECT: Both CS pumps are available. But all four CAR fans are not available. The "A" & "C" CAR fans will not be available because they will not have RBCCW to them. EOP 2525, SPTAs places the RBCCW pump on the facility without power to PTL when containment pressure is ≥ 20 psig. The facility 2 containment cooling equipment will be running. The "B" CS pump and "B" & "D" CAR fans have ESF actuation power (VA-20) and 4160 VAC power (24D). The "A" CS pump will be manually started once power is restored to 24C by closing the handswitch on C01.
PLAUSIBLE: The candidate may not remember that an RBCCW pump is not restarted after a loss of power when containment pressure is ≥ 20 psig.

D - INCORRECT: Both CS pumps and "B" & "D" CAR fans are available is correct. But power to 24C will be restored and the "A" CS pump started in EOP 2525 not 2532.
PLAUSIBLE: EOP 2525 is not written assuming the loss of VA-10. The design bases requires only one complete train of equipment in a design bases accident, therefore having only one train in EOP 2525 is sufficient.

SRO justification: Part of the question can be answered at the RO level. The RO should know that on a loss of VA-10 that actuation power is lost to facility 1 components. The RO will also have knowledge of what response not obtained actions are in EOP 2525. The SRO knowledge is how EOP 2525 will be implemented with a loss of VA-10. The crew is faced with multiple failures. The SRO must determine by knowledge of implementation strategy when available equipment will be started. EOP 2260 stated that the US should complete SPTAs prior to taking other actions. But that the US may direct actions beyond SPTAs, as necessary, to maintain Safety Functions, provided these actions do not result in a major change in plant conditions (e.g., equipment failures not included in the SPTAs). The SRO ensuring equipment ("A" containment spray pump) is started manually when it does not have actuation power is an action beyond SPTAs. No reference is provided. This question is linked to 10 CFR 55.43(b)(5).

Question #: **90** **Question ID: 2018036** **Rev.** 0 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/25/2018 1:35:19 PM ☐ RO ☑ SRO **Origin: New** ☐ Past NRC Exam?

References

EOP 2525, Standard Post Trip Actions
OP 2260, Unit 2 EOP User's Guide

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 022 022 (SF5 CCS) Containment Cooling

Number A2.04 **RO** 2.9* **SRO** 3.2 **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

A2.04 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: Loss of service water

Question #: **91** **Question ID: 1100132** Rev. 1 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/14/2018 8:42:41 AM ☐ **RO** ☑ **SRO** **Origin: Bank** ☐ Past NRC Exam?

The plant was at 100% when the power supply to the Plant Process Computer (PPC) was lost.
All the Initial Actions of AOP 2518, Loss of the Plant Process Computer, have been taken and the US is evaluating the results.

Which one of the following conditions would prevent continued operation at 100% power, due to administrative requirements that are affected by the loss of the PPC?

☐ **A** Proportional Heater Group #1 amp meter is inoperable.

☐ **B** Plant and site delta-T instruments are lost.

☑ **C** The "PWR RATIO HI/LO" annunciator on C-04 is in alarm.

☐ **D** Group 7 CEAs are not withdrawn to the UEL.

Question Misc. Info: MP2*LOIT, PPC, Loss of PPC, Audit-2011

Justification

A - WRONG; Monitoring of the PZR proportional hearers is required to ensure they are functioning according to the Tech. Spec. Requirements. Plant OE concerning PZR heater failures has necessitated continuous monitoring of the heater output by the PPC. When the PPC is lost, heater output must be manually calculated using the amp meters on C-04. However, the actual "administrative" requirements for this monitoring is 72 hours, not continuous.
PLAUSIBLE; Student may recall the heater monitoring requirements initiated by the past problems, but not remember these monitoring requirements are not as restrictive as the Tech Spec administrative requirements that would require a plant shutdown.

B - WRONG; Plant and site delta-T requirements of NPDES can be monitored by other means, although not as easily as with the PPC.
PLAUSIBLE; Student may remember AOP 2518 contains actions for loss of NPDES monitoring by the PPC (which are very strict during Winter Flounder Spawning season), but not remember the specifics of those required actions.

C - CORRECT; A functioning PPC is required to monitor the Incore Detectors, which are used to analyze core power distribution at various levels in the core. If they are unavailable and ASI is outside the required limits for the Excore Detectors, then Tech. Spec. 3.2.1 requires an immediate down power until ASI is within the required specification. The Power Ratio Hi/Lo annunciator being in alarm indicates ASI is <u>not</u> within its limits.

D - WRONG; Loss of PPC indication of the CEAs does require entry into a TSAS and based solely on this, would require entry into a power restrictive TSAS. However, AOP 2518 has steps to utilize an alternate pulse count indication to meet the TS requirements.
PLAUSIBLE; Student may consider the CEA reed indication on the Core Mimic as a "second" indication to replace the lost PPC pulse count rod position indication, which is allowed in Tech Specs for certain conditions. However, with Gp-7 not withdrawn to the Upper Electrical Limit (full out position), the Core Mimic is of no use. Therefore, a plant shutdown would be required within 24 hours per TSAS 3.1.3.3d

<u>SRO Justification</u>: This question is SRO only as it requires evaluation of plant conditions against the specific administrative action requirements of an abnormal procedure, in relation to the requirements of a specific Tech Spec Action Statement. The question cannot be answered by solely knowing systems knowledge, Technical Specification Safety Limits or LCO information listed in the LCO statement. This question is linked to 10 CFR 55.43(b)(5)

References

AOP 2518, Loss of Plant Process Computer; TSAS 3.2.1, Linear Heat Rate.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 016 016 (SF7 NNI) Nonnuclear Instrumentation

Number A2.02 **RO** 2.9* **SRO** 3.2* **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.5)

A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of power supply

Question #: **92** **Question ID: 2018037** **Rev. 0** ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/20/2018 10:17:13 PM ☐ **RO** ☑ **SRO** **Origin: Mod** ☐ Past NRC Exam?

Which one of the following is **NOT** required to have a current Tech Spec or TRM surveillance prior to fuel movement?

☐ **A** Refuel Machine load limit interlocks.

☐ **B** Source range speaker in containment.

☐ **C** Refuel station/control room communication link.

☑ **D** Upender/transfer carriage interlocks.

Question Misc. Info: MP2*LOIT NRC-2014

Justification

A - WRONG; TRM 3.9.6, Containment Building Refuel Machine surveillance requirements ensure the operation of the Programmable Logic Controller (PLC) on the Refuel Machine, which has several hoist load limits programmed in to prevent overloading the machine during various conditions. The absolute maximum limit is determined by external hardware is set to prevent exceeding the TRM limit of less than or equal to 3590 lbs in the "fuel plus hoist" region.
PLAUSIBLE; Student may consider the load limits of the refuel machine as being set by the manufacturer and are part of the machine's initial design criteria and would, therefore, only have to be verified following maintenance or modifications.

B - WRONG; Tech. Spec. 3.9.2 states the requirements for source range monitoring in containment and not necessarily on the Refuel Machine Bridge.
PLAUSIBLE: Because operators do NOT operate the Refuel Machine and do not perform any refueling surveillances, the student may believe the requirement is for the speaker that is normally used, which is located on the Refuel Machine itself. However, there is also a speaker on the wall inside CTMT that, if functioning, also meets the Tech. Spec. requirement.

C - WRONG; and TRM 3.9.5 states the requirements for the communication link to the control room. Both must meet surveillance requirements for fuel movement to take place.
PLAUSIBLE: Because the Refueling SRO only has to be in communication with the control room when a fuel bundle is actually being inserted or withdrawn from the core, but not necessarily when the Refuel Machine is being moved, the student may not believe the TRM surveillance requirements must be met before fuel movement is started.

D - CORRECT; TRM 3.9.6, Containment Building Refuel Machine surveillance requirements ensure the operation of the Programmable Logic Controller (PLC) on the Refuel Machine, which has several interlock functions to prevent fuel damage. However, there is NO surveillance requirements to ensure the Upender interlocks are functioning properly.

SRO Justification: This is a system question concerning refueling systems, which are SRO only systems. This question requires the SRO applicant to recognize a condition that should be prevented by interlock. No reference is provided. This question is linked to 10 CFR 55.43(b)(7).

References

TS 3.9.2, TRM 3.9.5, 3.9.6, 3.9.7

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 034 Fuel Handling Equipment System (FHES)

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 #: **N/A** **Question ID: 2014049** **Rev. 1** ☐ Student Handout? ☐ Lower Order?

☐ **RO** ☑ **SRO** **Origin: Parent** ☑ Past NRC Exam?

PARENT Which ONE of the following correctly describes
1. a refueling equipment design feature or interlock and
2. the Tech Spec LCO or TRM Requirement that is met by this feature/interlock.

☑ **A** (1) Control circuit design ensures lifting forces will not exceed refuel machine design maximum load capacity.
(2) TRM 3.9.6, Containment Building Refuel Machine

☐ **B** (1) An interlock on the SFP Platform Crane prevents manually placing a new fuel assembly in a Region C location.
(2) Tech Spec 3.9.18, Spent Fuel Pool - Storage

☐ **C** (1) An interlock prevents the SFP Platform Crane from colliding into the cask crane monorail hoist.
(2) TRM 3.9.7, Spent Fuel Pool Crane Travel Limits

☐ **D** (1) Control circuit design ensures the refuel machine will not operate unless the bridge-mounted source range speaker is energized.
(2) Tech Spec 3.9.2, Instrumentation

Question Misc. Info: MP2*LOIT NRC-2014

Justification

A - CORRECT: The Programmable Logic Controller (PLC) on the Refuel Machine has several hoist load limits programmed in to prevent overloading the machine during various conditions. The absolute maximum limit is determined by external hardware is set to prevent exceeding the TRM limit of less than or equal to 3590 lbs in the “fuel plus hoist” region.

B - Incorrect: There are TS restrictions for placing new fuel in Region C and the SFP Platform Crane is programmed to ensure proper placement of fuel bundles in the SFP; however, there are no interlocks to prevent manually placing new fuel in any SFP location.
Plausible: Operators do NOT operate the SFP Platform Crane resulting in unfamiliarity with the crane. Therefore, it would be logical to assume the new crane has an interlock to prevent placing new fuel in an area that Tech Specs would NOT allow.

C - Incorrect: TRM 3.9.7 provides travel limits to the Cask Crane to prevent fuel damage. There is NO interlock or design feature that prevents the SFP Platform Crane from colliding with the Cask Crane.
Plausible: An examinee may believe that an interlock prevents the collision which could result in fuel damage. Collisions have occurred before (OE).

D - Incorrect: There is a speaker on the Refuel Bridge dedicated to only providing audible indication of Source Range neutron counts. The speaker provides only audible signals and has NO interlocks or design features.
Plausible: Because the operators do not operate the Refuel Machine and the Refuel Machine is relatively new, it’s plausible to assume that a dedicated speaker could have an interlock to prevent Refuel Machine operation if the speaker is not energized.

SRO Justification: This is a system question concerning refueling systems, which are SRO only systems. This question requires the SRO applicant to recognize a condition that should be prevented by interlock.

Question References not yet listed.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 034 Fuel Handling Equipment System (FHES)

Number K4.02 **RO** 2.5 **SRO** 3.3 **CFR Link** (CFR: 41.7)

Knowledge of design feature(s) and/or interlock(s) which provide for the following: Fuel movement

Question #: **93** **Question ID: 2018038** Rev. 0 ☑ Student Handout? ☐ Lower Order?

Last Edited: 6/24/2018 11:21:13 AM ☐ RO ☑ SRO **Origin: New** ☐ Past NRC Exam?

The plant is in a refueling outage with fuel movement in progress.
The following additional conditions exist:

- Facility 1 is the protected facility.
- "B" Diesel Generator is out of service for planned maintenance and is due back in 5 days.
- All other plant systems and components are functioning as designed for the existing plant conditions.

Then, RM-9799A has a blown fuse and triggers annunciator C-01; C-40, "C.R.A.C.S. IN AUTO RECIRC. MODE".
The RO reports CRACS shifted into recirc mode and all applicable dampers closed on both facilities. However, Facility 1 filter fan did not start and will not manually start.
Maintenance reports CRACS Filter Fan, F-32A, needs a motor replacement which they can complete in 2 hours.

I&C and engineering report the blown fuse on RM-9799A was due to a manufacturing flaw in the control circuit board.
They state both CRACS rad monitors (RM-9799A & B) have the flaw and would not function as designed.

Which of the following describes the effect and required actions for the above conditions?

☐ **A** Both facilities of CRACS are inoperable unless Facility 2 is placed in service in recirc mode in 2 hours. All fuel movement must be temporarily halted until Facility 2 CRACS is in recirc mode.

☐ **B** Both facilities of CRACS are inoperable and 1 must be OPERABLE in 7 days or fuel movement must stop. Fuel movement may continue provided Facility 2 CRACS is placed in service in recirc mode.

☑ **C** Both facilities of CRACS are inoperable and Facility 2 must be placed in service in recirc mode in 1 hour. All fuel movement must be suspended until one facility of CRACS is made fully OPERABLE.

☐ **D** Both facilities of CRACS are inoperable and Facility 2 must be placed in service in recirc mode in 1 hour. Fuel movement may continue provided fan F-32A is returned to service within the 2 hours as stated.

Question Misc. Info: MP2 LOUT, MB-4829

Justification

A - WRONG; The I&C report indicates a "common mode failure" with both rad monitors. This would make both CRAC systems not Operable. But placing the Facility 2 CRAC system in service will not make it Operable but will allow fuel moovement.
PLAUSIBLE; Student may not consider the TSAS 3.3.3.1 and feel that based on the TSAS 3.7.6.1 required action to place a working facility of CRACS in recirc mode after 7 days, that doing so would meet the TSAS, allowing fuel movement.

B - WRONG; With Facility 1 CRACS inoperable and Facility 2 CRACS not having emergency power, fuel movement must immediately stop.
PLAUSIBLE; The student may believe the 7 day allowance of TSAS 3.7.6.1d.2 allows for fuel movement as long as the CRACS is in its accident mode.

C - CORRECT; The plant is in Mode 6 with irradiated fuel movement underway, so TSAS 3.7.6.1d and 3.7.6.1e apply. The ARP for C-01/C-40 (2590A-159) has a Caution that states, "To ensure proper cleanup of Control Room atmosphere, one complete *facility related train* of Control Room ventilation (i.e. supply, exhaust, and filter fans), must be in operation.". Because both rad monitors must be considered inoperable, TSAS 3.3.3.1b, Action 16-2 apply, which is to place Facility 2 CRACS in recirc mode within 1 hour. Because Facility 2 does not have an emergency power source, per TS 3.0.5 the actions of TSAS 3.7.6.1.e.2 also apply, which requires fuel movement be suspended until a facility of CRACS is made OPERABLE.

D - WRONG; TSAS 3.7.6.1e.1 states fuel movement must be immediately suspended due to both facilities of CRAC being not OPERABLE.
PLAUSIBLE; Student may believe the 2 hour requirement based on the lack of emergency power to Facility 2 CRACS is the key to maintaining one facility of CRACS functional.

SRO Justification: This question is SRO only as it requires evaluation of plant conditions as controlled by an Alarm Response Procedure AOP, against the specific administrative action requirements of multiple Tech Spec Action Statements. This includes TS 3.0.5 (not provided during exam), which deals with the effect of power supply loss on operability. The question cannot be answered by solely knowing systems knowledge, Technical Specification Safety Limits or LCO information listed in the LCO statement. This question is linked to 10 CFR 55.43(b)(2) and 10 CFR 55.43(b)(5)

Question #: **93** **Question ID: 2018038** **Rev. 0** ☑ Student Handout? ☐ Lower Order?

Last Edited: 6/24/2018 11:21:13 AM ☐ **RO** ☑ **SRO** **Origin: New** ☐ Past NRC Exam?

References **Provided**

Handout: TS 3.3.3.1 & TS 3.7.6.1
TS 3.0.5; ARP 2590A-159, R0C4 (Reference ONLY)

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 072 072 (SF7 ARM) Area Radiation Monitoring

Number A2.03 **RO** 2.7 **SRO** 2.9 **CFR Link** (CFR: 41.5 / 43.5 / 43.3 / 45.13)

A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the ARM system- and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Blown power-supply fuses

Question #: **94** | **Question ID: 2018040** | Rev. 0 | ☐ Student Handout? | ☑ Lower Order?

Last Edited: 6/19/2018 11:00:22 AM | ☐ RO | ☑ SRO | Origin: New | ☐ Past NRC Exam?

The crew performed a plant shutdown from 100% to 3% power in six hours. The RO was given a critical parameter to maintain reactor power between 3 to 4%.

A Work Control SRO, a couple hours after power was stabilize at 3%, identifies a discrepancy between delta T power and nuclear power. Delta T power is reading 2.5 % and stable and nuclear instrument power is reading 3X10-5% and lowering. RCS temperature and pressure have not changed since the plant was down powered.

Which instrument is indicating the power the reactor is producing AND what action must the onshift US direct?

☑ **A** Nuclear instruments are indicating the correct reactor power. The US will direct the reactor be shutdown.

☐ **B** Delta T power instruments are indicating the correct reactor power. The US will direct power be raised to 3 to 4 %

☐ **C** Nuclear instruments are indicating the correct reactor power. The US will direct power be raised to 3 to 4 %

☐ **D** Delta T power instruments are indicating the correct reactor power. The US will direct the reactor be shutdown.

Question Misc. Info: MP2*LORT/LOIT Fundamentals, CEA, reactivity

Justification

The Reactivity Management procedure OP-AA-300 specifies that whenever the status of reactor criticality becomes unknown, the reactor will be shutdown. With power on the nuclear instruments reading 3X10-5% the condition of the reactor is not known,i.e. whether the reactor is critical. The Reactivity Management procedure also states that it is the Unit Supervisors responsibility for ensuring Reactor Operators properly adhere to procedures, conservatively respond to abnormal reactivity events, and demonstrate proper respect for reactivity by using conservative operating practices. The safety and integrity of the core take precedence over power production.

This is OE for Surry in the early 2000s.

A - CORRECT: Nuclear instruments are indicating the correct reactor power. On the downpower xenon is building in and adding negative reactivity to the core. This will lower power. Delta T power has only lowered a little because it uses delta temperature to indicate power and the heat inputted to the RCS has not changed much because of the operating RCPs. Since the status of reactor criticality is unknown (3X10-5 %) the reactor must be shutdown.

B - INCORRECT: Delta T power instruments are indicating the correct reactor power is not correct. And the US will direct power be raised to 3 to 4 % is not correct because the condition of the reactor is not known.
PLAUSIBLE: Reactor power is just slightly outside the critical power band specified. The normal response would be to raise power and maintain it in the specified band. The candidate that does not understand why NI power and Delta T power are reading differently would probably select this answer.

C - INCORRECT: Nuclear instruments are indicating the correct reactor power is correct. But the US will direct power be raised to 3 to 4 % is not correct because the condition of the reactor is not known.
PLAUSIBLE: Nuclear instruments are indicating the correct reactor power is correct. The candidate that does not understand why NI power and Delta T power are reading differently and does not understand that at a power of 3X10-5 % the condition of the reactor is not known would probably select this answer. This is what occurred at Surry Nuclear power station in the early 2000s. The crew raised power from a very low level back to their pre event band.

D - INCORRECT: Delta T power instruments are indicating the correct reactor power is not correct since delta T power is indicating all heat input to reactor (such as RCP heat). And at lower power levels the pump heat is a significant input and indicates power higher that nuclear power. The US will direct the reactor be shutdown is correct.
PLAUSIBLE: A note in the Rapid Downpower procedure states that the PPC calorimetric may be inaccurate due to SG level transients and the most accurate available indication of reactor power is RPS ΔT power. And the US will direct the reactor be shutdown is correct.

SRO Justification: In this question the SRO assesses the facility condition and must use the conduct of operation procedure guidance to direct the appropriate action for an abnormal condition. No reference is provided. This question is linked to 10 CFR 55.43(b)(5).

Question #: **94** | ***Question ID:* 2018040** | **Rev.** 0 | ☐ **Student Handout?** | ☑ **Lower Order?**

Last Edited: 6/19/2018 11:00:22 AM | ☐ ***RO*** | ☑ ***SRO*** | ***Origin:* *New*** | ☐ **Past NRC Exam?**

References

OP-AA-300, Reactivity Management procedure
OP-2204, Load Changes

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

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

G2.1.7 Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation.

Question #: **95** **Question ID: 2018039** Rev. 0 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/19/2018 11:23:05 AM ☐ RO ☑ SRO Origin: New ☐ Past NRC Exam?

The Work Control SRO has directed a Plant Equipment Operator to change the position of a plant component. Which of the following is NOT an authorized method for changing the position of a plant component?

- ☐ **A** Approved Written Work Instructions
- ☐ **B** SM/US Direction
- ☑ **C** Standing Order
- ☐ **D** Engineering Design Change Process

Question Misc. Info: MP2*ILT, configuration control

Justification

OP-AA-1500, Operational Configuration Control, Attachment 5, specifies authorization to change the position of plant components is provided by one or more of the following:

- Approved procedure
- Approved tag-outs or equipment clearance
- Authorized work order
- Alternate Plant Configuration Sheet (Attachment 1), or equivalent
- Approved written work instructions
- Engineering design change process
- SM/US direction

A - INCORRECT: Approved Written Work Instructions is not correct since this is an approved method to change the position of a plant component.
PLAUSIBLE: Approved procedures and equipment clearances are the normal and most used methods to change the position of plant components. It is reasonable that a candidate could select this since it is not used very often.

B - INCORRECT: SM/US Direction is not correct since this is an approved method to change the position of a plant component.
PLAUSIBLE: Approved procedures and equipment clearances are the normal and most used methods to change the position of plant components. It is reasonable that a candidate could select this since there is not a piece of paper documenting the change.

C - CORRECT: Standing Order is correct. The Standing Order is not an approved method to change the position of a component specified in OP-AA-1500. Standing Orders are used to amplify or clarify operational information of a temporary nature; they are not used to change procedures or support operability, except in limited cases.

D - INCORRECT: Engineering Design Change Process is not correct since this is an approved method to change the position of a plant component.
PLAUSIBLE: Approved procedures and equipment clearances are the normal and most used methods to change the position of plant components. It is reasonable that a candidate could select this since Engineering Design Changes are implemented generally with authorized Work Orders. Rarely is just an Engineering Design Change used to authorize a change to a plant component.

SRO justification: This question is part of the SROs responsibility of maintaining plant configuration control using the facility's procedures for authorizing operating changes to the plant. No reference is provided. This question is linked to 10 CFR 55.43(b)(3).

References

OP-AA-100, Temporary (Shift and Standing Orders) Orders, R35
OP-AA-1500, Operation Configuration Control Challenge R15

NO Comments or Question Modification History at this time.

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)

G2.2.14 Knowledge of the process for controlling equipment configuration or status.

Question #: **96** **Question ID: 2018032** **Rev. 0** ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/8/2018 5:10:55 PM ☐ **RO** ☑ **SRO** **Origin: New** ☐ Past NRC Exam?

The plant is at 65% power and shutting down due to a recently discovered manufacturing defect in the bearings of both electric Auxiliary Feedwater Pumps, which rendered them inoperable.

Which of the following would administratively require the crew to stop the shutdown and maintain the plant in the existing Tech. Spec. Mode of operation, until the listed condition was corrected?

☐ **A** Generator voltage control is lost due to the loss of control power D-21.

☐ **B** The sudden loss of VR-21 due to a failure of its main breaker.

☐ **C** A CEA in Group 7 suddenly dropped to zero steps inserted.

☑ **D** The Turbine Driven Auxiliary Feedwater Pump is found to be inoperable.

Question Misc. Info: MP2*ILT TSAS, Tech. Spec., SRO

Justification

A - WRONG; The ARP (and subsequent AOP) states that if voltage control is lost, CONVEX must be immediately notified. Also, generator VAR loading must be verified to be within the normal range. However, it does not explicitly state that a power reduction can not be accomplished.
PLAUSIBLE; Student may recognize that this would make it impossible to control generator VARS as power was lowered, potentially damaging the main generator and causing grid instability.

B - WRONG; Although the loss of VR-21 would substantially complicate the operators control of the plant (either at power or in Mode 3), there is no administrative requirement to hold power stable until VR-21 is recovered.
PLAUSIBLE; Student may recognize that the loss of VR-21 would require the crew to isolate CVCS, removing their ability to borate the RCS to lower power. Also, it would trip all pressurizer backup heaters, preventing the crew from forcing PZR sprays and prevent the operation of the #2 ADV from the control room.

C - WRONG; Although AOP 2556, CEA Malfunctions says to hold power stable below 70% until the dropped CEA is recovered, this guidance is not meant to override actions required by other TSAS.
PLAUSIBLE; Student may remember the requirements for a dropped CEA and its recovery, and also recognize the fact that rods can not be used to control power with a misaligned CEA.

D - CORRECT; The Tech. Spec. 3.7.1.2 Auxiliary Feedwater Pumps, TSAS for all three AFW pumps being out of service is the only TSAS that <u>requires</u> the plant Mode of operation <u>not</u> be changed. The Bases for the Spec goes even further in stating that "the unit should not be perturbed by any action, including a power change that might result in a trip."

<u>**SRO Justification**</u>: This question is SRO only as it requires evaluation of changing plant conditions against the specific administrative action requirements of Tech Spec Action Statements and the requirements of the applicable TS Bases. The question cannot be answered by solely knowing systems knowledge, Technical Specification Safety Limits or LCO information listed in the LCO statement. This question is linked to 10 CFR 55.43(b)(2).

References

Tech. Spec. 3.7.1.2 Auxiliary Feedwater Pumps and its Bases.

NO Comments or Question Modification History at this time.

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 2.2.35 **RO** 3.6 **SRO** 4.5 **CFR Link** (CFR: 41.7 / 41.10 / 43.2 / 45.13)

G2.2.35 Ability to determine Technical Specification Mode of Operation.

Question #: **97** **Question ID:** **8500016** **Rev.** 5 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/13/2018 5:12:37 PM ☐ **RO** ☑ **SRO** **Origin:** **Mod** ☐ Past NRC Exam?

A LOCA Outside of Containment has occurred at the plant and conditions have degraded to a General Emergency event.

- The radiation release to the environment can be isolated by local valve operation.
- A qualified operator with no prior exposure for the year is available to perform the task.
- The area were these valves are located has a dose rate of 60 REM per hour.
- The task will take the individual 30 minutes to complete.

Which of the following describe the administrative requirements to complete the task of isolating the radiation release to the environment?

☑ **A** The individual must volunteer to perform the task; Only the DSEO can authorize the task.

☐ **B** They individual can be directed to perform the task; Only the DSEO can authorize the task.

☐ **C** The individual must volunteer to perform the task; Only the MCRO can authorize the task.

☐ **D** They individual can be directed to perform the task; Only the MCRO can authorize the task.

Question Misc. Info: MP2*LOIT Emerg Rad Exposure, ALARA, NRC-2008 [K/A; 2.3.4], NRC-2016, Audit-2016

Justification

A - CORRECT; This equates to a dose of > 25 Rem, which exceeds the Emergency Exposure Limit for non-volunteers performing accident mitigation. Therefore, they have to volunteer and only the DSEO can authorize the task and this cannot be delegated.

B - WRONG; This is the correct person who must authorize the task, but over 25 rem of exposure must be voluntary.
Plausible; Student may feel that because protecting the general population from potentially serious radiation exposure is our job, this level of exposure and can be directed.

C - WRONG; This is correct for the person performing the task, but any exposure that could exceed the emergency limit of 25 rem must be authorized by the DSEO directly.
Plausible; Student may believe that any plant manipulation must be the sole responsibility of the Shift Manager, who is the senior licensed individual on watch and is ultimately responsible for the accident mitigation.

D - WRONG; If the task will cause an exposure that will exceed 25 rem, the individual must be a volunteer and only the DSEO can authorize the task.
Plausible; Student may consider protecting the welfare of the general public as part of the job expectation of any nuclear operator and the SM is ultimately responsible to ensure the accident is mitigated such that this is accomplished.

SRO Justification: This question is SRO only as it requires an understanding of the specific administrative requirements of the Emergency Plan Implementation Procedures and the responsibilities of Station Emergency Response Organization personnel at the Director or Assistant Director level. The question cannot be answered by solely knowing systems knowledge, Technical Specification Safety Limits or LCO information listed in the LCO statement. This question is linked to 10 CFR 55.43(b)(4).

References

EPIP FAP09, R5C0, Attachment 2 table.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.3 Radiation Control

Generic K/A Selected

NRC K/A Generic **System** 2.3 Radiation Control

Number 2.3.4 **RO** 3.2 **SRO** 3.7 **CFR Link** (CFR: 41.12 / 43.4 / 45.10)

G2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.

Question #: **N/A** **Question ID: 8500016** Rev. 3 ☐ Student Handout? ☐ Lower Order?

☐ **RO** ☐ **SRO** **Origin: Parent** ☑ Past NRC Exam?

PARENT

A LOCA Outside of Containment has occurred at the plant. In addition, excessive fuel damage has resulted in radiation levels significantly above normal in various equipment locations. A General Emergency Classification has been declared and the Station Emergency Response Organization is fully staffed.

It was determined that the LOCA could be isolated by manual valve manipulation in an area where the dose rates are approximately 60 REM per hour. An operator has entered the area and isolated the leak, but appears to have suffered a stroke and now needs assistance to leave the high radiation area.

ALL dose extensions necessary for this situation have been granted per the Emergency Exposure Limits guidelines and documented on a Radiation Work Permit that covers the applicable high radiation area.

Which of the following exposure requirements would still be applicable for the individual (with no prior exposure for the year) who enters the high radiation area to assist the injured operator?

☐ **A** Any male or non-pregnant female can volunteer; stay time is limited to 6 minutes.

☐ **B** Only males over 50 can volunteer; stay time limit is 25 minutes.

☐ **C** Only males any age can volunteer; stay time is up to that individual.

☑ **D** Any male or female can volunteer; stay time is up to that individual.

Question Misc. Info: MP2*LOIT Emerg Rad Exposure, ALARA, NRC-2008 [K/A; 2.3.4], NRC-2016

Justification

A - WRONG; This equates to a dose of 5 rem, which is the normal limit for non-emergency scenarios.
Plausible; Examinee may recognize that any employee can volunteer, but use the normal exposure limits to calculate the stay time.

B - WRONG; This equates to a dose of 25 Rem, which is the Emergency Exposure Limit for non-volunteers performing accident mitigation. The "male over 50" is a company guideline when soliciting volunteers for high exposure missions, but it is NOT a requirement.
Plausible; Examinee may use the "accident mitigation" exposure limit as the highest allowed for any circumstance.

C - WRONG; This is the correct dose for life-threatening emergency situations. However, although excluding females is plausible, it is NOT an administrative requirement.
Plausible; Examinee may consider the "male over 50" company guideline as an actual requirement.

D - CORRECT; For "life saving situations" the dose limit per Emergency Exposure Limits is strictly up to the individual who volunteers to give assistance. In this instance, procedures do NOT have different requirements for males or females as the person must be a volunteer.

Question References not yet listed.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.3 Radiation Control

Generic K/A Selected

NRC K/A Generic **System** 2.3 Radiation Control

Number 2.3.7 **RO** 3.5 **SRO** 3.6 **CFR Link** (CFR: 41.12 / 45.10)

Ability to comply with radiation work permit requirements during normal or abnormal conditions.

Question #: **98** **Question ID:** **87989** **Rev.** 2 ☐ Student Handout? ☐ Lower Order?

Last Edited: 6/8/2018 6:36:33 PM ☐ **RO** ☑ **SRO** **Origin:** **Bank** ☐ Past NRC Exam?

Prior to moving irradiated fuel in the reactor vessel, the reactor must have been subcritical for a minimum period of time.

This delay assures which of the following?

☐ **A** Sufficient radioactive decay of N-16 in the Reactor Coolant System.

☐ **B** Heat from the spent fuel will NOT exceed the Spent Fuel Pool Cooling capacity.

☑ **C** Sufficient radioactive decay of the short-lived fission products.

☐ **D** Fuel movement will NOT result in criticality in the Spent Fuel Pool.

Question Misc. Info: MP2*LOIT Refuel - Tech Spec 3.9.3.1, basis for waiting 100 hours

Justification

A - WRONG; This is the bases for containment entry requirements.
PLAUSIBLE; Student may remember the limit for CTMT entry and believe it is the basis for the TS.

B - WRONG; This is the bases for the maximum amount of core offload allowed.
PLAUSIBLE; Student may remember there is a definite limit to the heat removal capacity of the SFPC system and recognize that decay heat dissipation would play a big part in staying under the limit.

C - CORRECT; This is the bases for 3.9.3.1, Decay Time, so that the calculated radiological dose consequences of the fuel handling accident are bounding.

D - WRONG; This is the basis for 3.9.1, Boron Concentration.
PLAUSIBLE; Student may remember the bases relates to short-lived fission products but believe this is more of a reactivity concern.

SRO Justification: This question is SRO only as it requires knowledge and understanding of the applicable TS and its Bases. The question cannot be answered by solely knowing systems knowledge, Technical Specification Safety Limits or LCO information listed in the LCO statement. This question is linked to 10 CFR 55.43(b)(2).

References

Tech Spec basis 3/4.9.3

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.3 Radiation Control

Generic K/A Selected

NRC K/A Generic **System** 2.3 Radiation Control

Number 2.3.12 **RO** 3.2 **SRO** 3.7 **CFR Link** (CFR: 41.12 / 45.9 / 45.10)

Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Question #: **99** **Question ID:** **88807** **Rev.** 1 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/14/2018 8:44:26 AM ☐ **RO** ☑ **SRO** **Origin:** **Bank** ☐ Past NRC Exam?

The plant has tripped from 100% power due to a small Steam Generator Tube Rupture on the #1 SG.
On the trip, offsite power was lost. Auxiliary Feedwater is unavailable due to a pipe rupture at the AFW Reg. Valves.
All other plant systems and components are functioning as designed.
The crew has transitioned to EOP 2540, "Functional Recovery" and is assessing the course of action.

The following conditions now exist:

- RCS pressure = 1550 psia and stable.
- RCS temperature = 532°F and stable.
- Pressurizer level = 20% and slowly dropping.
- Reactor Vessel level = 100%.
- #1 SG level = 35% and slowly rising.
- #2 SG level = 90" and dropping.

Which of the following EOPs must be entered and what is the basic directed action for RCS cooldown/depressurization?

☐ **A** EOP 2540C1, Recovery of RCS Inventory, IC-2, Safety Injection flow;
Fully open ADVs and open PORVs to depressurize.

☐ **B** EOP 2540D, Recovery of Heat Removal, HR-2, SG Heat Sink with SI Operating;
Throttle open ADVs and use Main/Aux. Spray to depressurize.

☐ **C** EOP 2540E, Recovery of Containment Isolation, Appendix 12, SGTR Response;
Throttle open ADVs and use Main/Aux. Spray to depressurize.

☑ **D** EOP 2540D, Recovery of Heat Removal, HR-3, Once-Through-Cooling;
Fully open ADVs and open PORVs to depressurize.

Question Misc. Info: MP2*LOIT E40-01-C, 2540, EOP, SRO, NRC, K/A: E09/2.2.44

Justification

A - WRONG; RCS Heat Removal is the key Safety Function not being met.
PLAUSIBLE; Student may consider RCS Inventory Safety Function is the major concern because SI flow will maintain SG level for Heat Removal. Therefore, SI flow is required for both Inventory and Heat Removal and RCS or SG pressure is holding up injection flow.

B - WRONG; These are the contingency actions if Heat Removal were not being met due to high RCS pressure.
PLAUSIBLE; Student may recognize RCS pressure is blocking SI flow (only charging pumps are injecting), and this would be the correct action if #1 SG level was a result of feed flow.

C - WRONG; The CTMT Isolation Safety Function is not met, but Heat Removal is a higher SF.
PLAUSIBLE; Student may recognize the SGTR will cause the CTMT Isolation Safety Function to not be met and consider this the critical action due to the ongoing radiation release.

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

SRO Justification: This question is SRO only as it requires knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures. The question cannot be answered by solely knowing systems knowledge, Technical Specification Safety Limits or LCO information listed in the LCO statement. This question is linked to 10 CFR 55.43(b)(5)

References

EOP 2540-002, R4C0, Resource Assessment Trees, HR-3.

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Question #: **99** | ***Question ID:*** ***88807*** | **Rev.** **1** | ☐ **Student Handout?** | ☑ **Lower Order?**

Last Edited: 6/14/2018 8:44:26 AM | ☐ ***RO*** | ☑ ***SRO*** | ***Origin:*** ***Bank*** | ☐ **Past NRC Exam?**

Number 2.4.1 **RO** 4.6 **SRO** 4.8 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

G2.4.1 Knowledge of EOP entry conditions and immediate action steps

Question #: **100** **Question ID: 2016021** Rev. 2 ☐ Student Handout? ☑ Lower Order?

Last Edited: 6/14/2018 8:44:44 AM ☐ RO ☑ SRO **Origin: Mod** ☐ Past NRC Exam?

A fire has occurred in the 25'6" Auxiliary Building Cable Vault. Fire suppression activities appear to be causing uncontrolled component operations in the plant. #1 MSIV, MS-64A, suddenly closes, causing a plant trip.

The SM orders a Control Room evacuation and control of the plant be restored using AOP 2559, "Fire" and AOP 2579A, "Fire Procedure for Hot Standby Appendix R Fire Area R-1".

In accordance with C OP 200.18, "Time Critical Action Validation And Verification", which of the following actions must be completed in <u>less than two minutes from the time of trip</u>, due to the SM classifying the fire an Appendix R Fire Area R-1?

☐ **A** The Security Shift Supervisor must be notified of the request for offsite assistance (Fire and Police).

☐ **B** The Main Steam Isolation Valves (MSIVs) must be isolated at the Bottle-Up Panels following a plant trip.

☑ **C** The Power Operated Relief Valves (PORVs) must be isolated at the Bottle-Up Panels following a plant trip.

☐ **D** Control of Letdown Isolation valve, 2-CH-089, must be transferred to the Fire Shutdown Panel, C-10.

Question Misc. Info: MP2*LOIT Fire, App. "R", AOP, 2559, TS, TSAS, Bases, SRO, NRC-2016 [Vision# 386495]

Justification

A - WRONG; This action is done in the initial steps of AOP 2559, but is NOT considered a Time Critical action.
PLAUSIBLE; Student may recall the procedural requirement to notify Security of the need for offsite assistance, and consider the potential impact of delaying their arrival due to station security requirements as a constituting a Time Critical Action,

B - WRONG; The MSIVs are required to be isolated at the Bottle-Up panels within 3 minutes of the plant trip.
PLAUSIBLE; Student may remember the MSIVs are required to be isolated at the Bottle-Up panels, but not that they are the second on the list behind the PORVs because a plant trip is required for the MSIVs.

C - CORRECT; The AOP 2559 requires the switches on the Bottle-up Panels be placed in isolate for the PORVs, MSIVs and ADVs. In the case of the PORVs, they must be isolated **within 1 minute 55 seconds** of a plant trip caused by MSIV closure to prevent an unacceptable loss of RCS pressure control.

D - WRONG; Control of 2-CH-089 must be transferred to C-10 within five minutes of a plant trip.
PLAUSIBLE; Student may remember that control of the letdown isolation valve must be done expeditiously, but not when.

<u>**SRO Justification**</u>: This question is SRO only because it requires knowledge of an action that relates to an SRO only <u>directed</u> action (not an Immediate Operator Action) based on the specific fire area and the specific cause of the plant trip. The question cannot be answered by solely knowing systems knowledge, Technical Specification Safety Limits or LCO information listed in the LCO statement. This question is linked to 10 CFR 55.43(b)(5).

References

AOP 2559, R11C0, Fire, St. 5.2;
AOP 2579A, R12C0, Fire in Area R-1, Att 17

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.4.11 **RO** 4.0 **SRO** 4.2 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

G2.4.11 Knowledge of abnormal condition procedures.

Question #: **N/A** **Question ID: 2016021** Rev. 0 ☐ Student Handout? ☑ Lower Order?

☐ RO ☑ SRO **Origin: Parent** ☑ Past NRC Exam?

PARENT

A fire has occurred in the 25'6" Auxiliary Building Cable Vault. Fire suppression activities appear to be causing uncontrolled component operations in the plant. The SM orders a Control Room evacuation and control of the plant be restored using AOP 2559, "Fire" and AOP 2579A, "Fire Procedure for Hot Standby Appendix R Fire Area R-1".

Which of the following plant components would be rendered not OPERABLE or capable of performing its design function, once control of the plant is regained per AOP 2579A?

- ☑ **A** Power Operated Relief Valves (PORVs).
- ☐ **B** Main Steam Isolation Valves (MSIVs).
- ☐ **C** Atmospheric Dump Valves (ADVs).
- ☐ **D** Auxiliary Feed Regulating Valves (AFRVs).

Question Misc. Info: MP2*LOIT Fire, App. "R", AOP, 2559, TS, TSAS, Bases, SRO, NRC-2016

Justification

SRO Justification; 10CFR55.43(1) SRO needs to understand the Conditions and Limitation of the facility license to determine OPERABILITY for components knowing the required Tech. Spec.

A - CORRECT; The AOPs require the switches on the Bottle-up Panels be placed in isolate for the PORVs, MSIVs and ADVs. This will fail closed both facilities of all three of these components. It also requires all control power except VA-20 be deenergized, failing open the Fac. 1 AFRV. However, IAW Tech. Spec. Bases, the PORVs are suppose to be available for RCS pressure control if needed.

B - WRONG; The MSIVs are in their accident position when closed.
Plausible; Procedurally required when evacuating the control room for a fire, the operators are required to place all isolation switches in the Bottle Up Panel to “ISOLATE” therefore for both MSIV the, Examinee may determine this action INOPs the valves.

C - WRONG; The ADVs are not required to be operated remotely from the control room to perform their design function.
Plausible; Procedurally required when evacuating the control room for a fire, the operators are required to place all isolation switches in the Bottle Up Panel to “ISOLATE” , which isolates all control signals to the "A" ADV and isolates all control signals to the "B" ADV except those from C-10. Therefore, the Examinee may determine that this action INOPs both ADVs.

D - WRONG; The AFRVs are in their accident position when failed open.
Plausible; Procedurally required to evacuate to Fire Shutdown Panel C-10 the Examinee may think that since there are no controls for Facility 1 Aux Feed Reg Valve that this would make the Valve not OPERABLE.

References

AOP 2559

NO Comments or Question Modification History at this time.

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

Number AA2.14 **RO** 3.2 **SRO** 4.3 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Plant Fire on Site: Equipment that will be affected by fire suppression activities in each zone