Question #: 1 | **Question ID; 1000110** | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

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

A reactor trip occurred from 100% power. During the performance of EOP 2525, Standard Post Trip Actions, the BOP notes that both 6900 volt AC buses are deenergized due to a failure to transfer to the RSST. All other electrical buses are energized from their normal source.

Which of the following describes the effect of the loss of power on the secondary system and the appropriate action to be taken?

☐ **A** The loss of condensate flow to the Gland Exhaust Condenser will cause a loss of condenser vacuum. Per EOP 2526, Reactor Trip Recovery, the BOP must immediately open MS-182A, Gland Seal supply bypass.

☑ **B** Condenser vacuum will be lost due to the loss of cooling to the steam jet air ejector. Per EOP 2525, Standard Post Trip Actions, the BOP must close the MSIVs and open 2-AR-17, Condenser Vacuum Breaker.

☐ **C** Condenser hotwell level will rapidly rise due to the loss of condensate flow. Per OP 2525, Standard Post Trip Actions, the BOP must limit Auxiliary Feedwater flow to less than 300 gpm to each steam generator.

☐ **D** The condensate surge tank will overflow resulting in a chemical release to the environment. Per OP 2319B, Condensate Storage and Surge System, the BOP must close the reject valve to the surge tank.

Justification

B - Correct; A loss of 6900 Volt AC buses results in a loss of condensate pumps. The loss of condensate will result in a loss of condenser vacuum due to the loss of cooling for the SJAE and the loss of flow to the feed pump seals. EOP 2525, step 11 states that if offsite power is lost or the condenser is NOT available, then close both MSIVs and open 2-AR-17, condenser vacuum breaker.
A - Wrong; the turbine seals will still be maintained by gland seal steam until the MSIVs are closed. The gland seal supply valves will automatically throttle to maintain the appropriate steam seal pressure.
C - Wrong; On a loss of condensate pumps, the subsequent actions of EOP 2525 require the BOP to close the MSIVs. With NO steam flow to the condenser, hotwell level will NOT rise; therefore Auxiliary Feed flow is NOT limited to 300 gpm. Aux Feed flow is limited only to prevent overfeeding S/Gs and is NOT based on hotwell level.
D - Wrong; because condensate pump discharge pressure provides flow to the CST. A loss of 6900 volt AC buses causes a loss of condensate pumps; therefore the CST will NOT fill up.

References

LP CAR-00-C, R-3, C-5, Condenser Air Removal, (three linked locations)
EOP-2525, R-23, St. 10 (Inst & Cont)

Comments and Question Modification History

Distractor A: Replaced ". .Gland Exhaust unloading valve." with ". .MS-182A, Gland Seal supply bypass." 11/11/08
Distractor C: Was a correct statement and was replaced on 11111108. "Severe water hammer will occur in the feedwater heaters due to the loss of cooling by condensate. Per EOP 2525, Standard Post Trip Actions, the BOP must open 2-HD-106, subcooling valve."

NRC K/A System/E/A System E02 Reactor Trip Recovery

Number EA1.l RO 3.7 SRO 3.7 **CFR** Link (CFR: 41.7 145.5 145.6)

Ability to operate and/or monitor components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features as they apply to the Reactor Trip Recovery.

Question #: **2** | ***Question ID:*** ***8000001*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ Student Handout? | ☐ Lower Order?

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

The plant had tripped from 100% power due an inadvertent MSI on Facility1

On the trip, one Pressurizer Safety valve opened and remained open. All other plant equipment responded as designed. The crew completed EOP 2525 and has transitioned to the applicable event specific EOP.

The BOP has commenced a plant cooldown by opening both Atmospheric Dump Valves (ADVs) 50%, using the controllers on C-05 in manual mode.

However, the #1 ADV valve operator is failed such that the valve is only open 25%.
The BOP believes, based on plant response, that the #1 ADV is open significantly less than 50% and wants to verify the actual position of the valve.

Which one of the following describes the method the BOP must use to verify the actual position of #1 ADV.

☐ **A** EITHER the #1 ADV's "red" light on C05 or the C05 "ADV Not Closed" annunciator.

☐ **B** EITHER the Plant Process Computer or the Foxboro IA Steam Dump Control screens.

☑ **C** ONLY local valve position indication on the valve.

☐ **D** ONLY the Foxboro IA Steam Dump Control screens.

Justification

C - Correct; Unlike the Condenser Steam Dump Valves, the #1 ADV operator has NO remote feedback as to actual valve position Therefore, it can ONLY be verified locally.
A - Wrong; A limit switch operated by valve stem movement energizes the "red" light on C05 when the valve is about 15% open. Each ADV has an individual "red" light on C05, however either limit switch moving about 15% will actuate the common " ADV Not Closed" annunciator on C05.
B - Wrong; The PPC displays the indication sent to it by the Foxboro IA system, which is DEMAND indication, NOT actual valve position.
D - Wrong; Only the Condenser Steam Dump Valves have an actual valve position detector built into each valve positioner, which feeds back to the Foxboro IA and is relayed on to the PPC for display on both.

References

Loss-Of-Control-Power Operator Aid, R-1, C-0
LP ESA-01-C, Engineered Safety Features Actuation System, Pg. 19
One-Line Diagram of Steam Dump/Turbine Bypass Control System, CL242

Comments and Question Modification History

Rephrased the stem of the question to shorten it and to make it less confusing. 11111108

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

Number AA2.17 RO 2.5 SRO 2.7* CFR Link (CFR: 43.5 145.13)

Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: Steam dump valve controller (position)

Question #: 3 | **Question ID: 8000045** | ☑ **RO** | ☐ **SRO** | ☐ Student Handout? | ☐ Lower Order?

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

The plant was at 20% power, downpowering for a refueling outage, when the following occurred:
- The plant was tripped due to a rupture of the charging header somewhere near the RCS loop.
- Two CEAs stuck out (fully withdrawn) on the trip.
- The Main Steam Header ruptured in the Aux. Feed Pump Room, on the steam supply to the Turbine Driven Aux. Feed Pump.

Present conditions 15 minutes post trip are:
- The crew has entered EOP-2540 for a Small Break LOCA and a Steam Line Break.
- Reactor power is stable at ~ 5 E- 05%
- RCS pressure dropped to, and is stable at ~ 1500 psia.
- RCS Tavg is stable at ~ 460°F
- The charging header has been isolated from the control room.
- Ruptured steam supply to the TDAFP has been isolated.
- SIAS, CIAS, EBFAS and MSI have been verified as fully actuated.
- All other plant equipment and conditions are as expected for this event.

The Reactivity Control Safety Function is ______________________

☐ **A** NOT met based on two CEAs fully withdrawn and no boron injection.

B NOT met based on little or no boron injection or Xenon production.

☑ **C** presently met based on the existing reactor power level and trend.

☐ **D** presently met based on SIAS boron injection and stable RCS Tavg.

Justification

C - Correct; EOP-2540 Reactivity Control Safety Function requires a reactor power level at or below 1 X 10 -04% and stable to meet "Condition 2" of the acceptance criteria.
A -Wrong; This is the requirement of EOP-2525 for more than one CEA not fully inserted on the trip. However, EOP-2540 looks for a shutdown reactor based on actual power level.
B - Wrong; There would have been little or no boron injection and not enough time for any appreciable Xenon production. However, these criteria do not apply for the Reactivity Control Safety Function. They would come into play during accident recovery much later.
D -Wrong; If either accident had lowered RCS pressure ~ 250 psi, this condition would help ensure the reactor remained shutdown. However, there is no SI flow due to RCS pressure and the charging header rupture.

References

EOP-2540A, R-10, St. 3, Condition 2 (Inst)
EOP-2541, R-2, App. 2, Figure 3 - Pre-SRAS Minimum Required SI Flow

Comments and Question Modification History

Revised stem to be a 'fill in the blank.' Revised answers to allow each of them to complete the sentence. 11111108

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

Number EA2.32 RO 3.2* SRO 3.6* CFR Link (CFR 43.5145.13)

Ability to determine or interpret the following as they apply to a small break LOCA: SDM

Question #: **4** | **Question ID: 8000002** | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | **Origin:** New | ☐ Past NRC Exam?

The plant has tripped due to a Large-Break LOCA and the crew has been successfully mitigating the event using the applicable EOP.
The following plant conditions exist six hours into the event:

- SIAS, CIAS, CSAS, MSI and SRAS have all been verified as completely actuated.
- CTMT pressure = 2.3 psig and lowering slowly.
- CTMT temperature = 220°F and lowering slowly.
- Reactor Vessel Level = 19% and stable.
- All other plant equipment is functioning as designed.

Then, at that time, a state wide blackout causes a loss of the RSST and the following conditions now exist:

- Facility 1 components are unavailable due to an electrical fault on bus 24C and/or 24E (24E is presently aligned to 24C).
- "B" EDG starts and its output breaker closes but the Fac. 2 Sequencer has failed and does NOT re-start any components.

Which of the following lists pumps that MUST be operating and why?

A Service Water, RBCCW and HPSI pumps for core cooling.
Service Water and RBCCW pumps and CAR Fans for CTMT cooling.

☐ **B** Aux Feedwater pumps for RCS cooling.
Service Water, RBCCW and HPSI pumps for core and CTMT cooling.

☐ **C** Service Water, RBCCW and LPSI pumps for core cooling.
Service Water, RBCCW, and CTMT Spray pumps for CTMT cooling.

☐ **D** Service Water and RBCCW and LPSI pumps for RCS cooling.
CTMT Spray pump and CAR Fans for CTMT and core cooling.

Justification

A - Correct; SW is the heat sink to RBCCW. RBCCW is the heat sink to the RCS because vessel level is too low to use the SGs as a heat sink. HPSI is required for flow through the core because LPSI cannot be used during Sump Recirc. Present CTMT pressure and time dictate that CAR fans be used for CTMT cooling, not CTMT Spray.
B -Wrong; The CAR fans are utilized by procedure at this time. With vessel level at 19%, the SGs are not used for RCS cooling, therefore there is NO need for Aux. Feed Water. EOPs dictate HPSI injection is used.
C - Wrong; in sump recirc (SRAS) the LPSI pumps are not used for RCS or core cooling.
D - Wrong; With CTMT pressure less than 7 psig, CTMT spray would be secured by procedure.

References

EOP-2532, St. 5, 13, 22, 23 and 60

Comments and Question Modification History

Reworded all four choices to reduce number of words and Improve readability.

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

Number EK3.03 **RO** 4.1 **SRO** 4.3 **CFR Link** (CFR 41.5 / 41.10 145.6 / 45.13)

Knowledge of the reasons for the following responses as the apply to the Large Break LOCA: Starting auxiliary feed pumps and flow, ED/G, and service water pumps

Question #: **5** | **Question ID: 8054462** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

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

Assume RCP seal flow for the "B" RCP reaches 10 gpm and its bleedoff excess flow check valve closes.

Which one of the following describe the ~~undesirable effect~~ Consequence of this valve closing?

A Bleedoff flow from other RCPs will be blocked, potentially damaging the seals in the other three RCPs.

☑ **B** The "B" RCP must be secured due to the loss of seal flow and the potential for failure of the RCP seal.

☐ **C** All RCP bleedoff flow must be diverted to the PDT to prevent exceeding temperature and pressure limits.

☐ **D** The bleedoff line relief valve will lift, resulting in excessive leakage of the RCS to the containment.

Justification

B - Correct; If bleedoff flow to an RCP seal is blocked, it would cause damage to the seals. The applicable plant procedures require the RCP be immediately secured, even if it involves a required plant trip.
A - Wrong; The purpose of the Excess Flow Check valve is to prevent the blocking of bleedoff flow from all RCPs due to high bleedoff flow from one RCP.
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.
D - Wrong; This would only occur if both the CTMT isolation valve and the PDT isolation valve were to close. A possible scenario if the bleedoff check valve did NOT seat and flow was so high it exceeded the operator's ability to pump the PDT.

References

ARP for Control Board alarm CA-21, "RCP 'B' Bleedoff Flow Lo", ARP-25908-101, R-0, C-1, St. 4 & 5

Comments and Question Modification History

Added 'All' to the beginning of distractor C to provide clarification and to make the distractor more plausible. (Cannot align any individual RCP bleedoff to the PDT) 11/11/08

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

Number AK2.07 **RO** 2.9 **SRO** 2.9 **CFR Link** (CFR 41.7 145.

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

Question #: **6** | ***Question ID:* 8000004** | ☑ ***RO*** | ☐ ***SRO*** | ☐ Student Handout? | ☑ Lower Order?

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

The plant is at 100% power, steady state, when the rupture of a charging pump discharge dampener requires charging and letdown be secured.

How long can the plant stay on line before administrative requirements necessitate a plant trip?

☑ **A** Until pressurizer level lowers to 10% below level setpoint (55% actual).

☐ **B** Until pressurizer level drops below the Tech. Spec. limit of 35%.

☐ **C** Until pressurizer level drops below the heater cutout limit of 20%.

☐ **D** Until lowering pressurizer level causes RCS pressure to drop to 2225 psia.

Justification

A - Correct; AOP-2512, Loss Of All Charging, requires the plant be tripped if pressurizer level drops to 10% or more below pressurizer level setpoint. This level will be reached long before the pressurizer lowers to the values stated in any of the distracters.
B -Wrong; The Tech. Spec. limit of 35% requires action and eventual plant shutdown, but NOT a plant trip.
C - Wrong; When pressurizer level reaches 20% all heaters will automatically de-energize, and there is an administrative requirement to shutdown upon loss of the Proportional Heaters, but NOT a plant trip.
D - Wrong; This is the DNB RCS pressure limit, which requires entry into a Tech. Spec. Action Statement and a plant shutdown, but NOT a plant trip.

References

AOP-2512, R-1, C-2, Note preceding St. 3.1 and St. 3.1 (Inst)

Comments and Question Modification History

Modified Answer 'A' slightly to provide clarification. Was "Until pressurizer level lowers to 10% or more below programmed level."
1111108

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

Number AA2.04 RO 2.9 SRO 3.8 CFR Link (CFR: 43.5 145.13)

Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Pump Makeup: How long PZR level can be maintained within limits

Question #: **7** | ***Question ID:*** ***8054131*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The plant is on Shutdown Cooling using the " A LPSI pump. The "B" LPSI pump is available but is NOT in service because of a Loss of Normal Power (LNP) surveillance that is scheduled to be performed on Facility 2. The LNP surveillance involves opening up the RSST supply breaker to bus 24D, in order to test the Emergency Diesel Generator (EDG) and the ESAS Sequencer's ability to restore specific vital loads.

Immediately after the LNP test is initiated, the " A LPSI pump breaker trips on overload due to a seized impeller.

Which one of the following describes the automatic response of the "B" LPSI Pump once the ESAS Sequencer completes restarting all applicable loads?

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

☐ **B** Because the "B" LPSI pump handswitch is in the "Neutral" position, it will automatically restart the moment 24D is reenergized.

☐ **C** Because the "B" LPSI pump is aligned for SDC use, it will automatically restart on the applicable sequencer step.

☐ **D** Because there is NO SIAS present on ESAS, the "B" LPSI pump must be manually restarted once 24D is reenergized.

Justification

D - Correct; There is no automatic restart of the LPSI pumps on an LNP, hut they do aet a load shed signal.
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.
B - Wrong; The LPSI Pump hand switch position does NOT affect the LNP start signal. However, this does require the operator place the handswitch to the stop position before attempting to restart the pump.
C - Wrong; When the plant enters the shutdown mode, plant procedure steps jumper out some of the control interlocks that would automatically trip the LPSI pumps on certain switchyard (grid) events. The only "automatic" restart for a LPSI pump is from a SIAS. These signals are not triggered by the LNP surveillance, nor do the control jumpers have any impact on the LNP load shed of the LPSI pumps.

References

LP SDC-00-C, R-4, Pg. 12; Causes of SDC pump trip and effect of LNP on running pump

Comments and Question Modification History

Reworded the stem to change the order of the LNP test and the loss of the " A LPSI Pump. The Facility 2 LNP test would NOT continue if the "A LPSI Pump were lost immediately before the test. 11/11/08

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

Generic KIA Selected

NRC **KIA Generic** System 2.2 Equipment Control

Number 2.2.12 RO 3.7 SRO 4.1 CFR Link (CFR: 41.10 145.13)
Knowledge of surveillance procedures.

Question #: 8 | **Question ID: 8000006** | ☑ RO ☐ SRO | ☑ Student Handout? | ☑ Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

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

Then, the " A RBCCW pump suddenly trips on overload.

Which one of the following plant conditions would require ADDITIONAL notifications to be made to internal or external personnel?

☑ **A** If the "B" RBCCW pump is not started within five (5) minutes, the plant must be tripped and plant personnel must be notified of the trip by the plant paging system.

☐ **B** If one restart of the " A RBCCW pump has been authorized by the SM, the Auxiliary Building PEO must be notified to check the " A pump breaker before restarting the pump.

☐ **C** If the "B" RBCCW pump is available, but 24E is aligned to 24D, senior plant management must be notified before allowing the Facility 2 pump to used on Facility 1.

☐ **D** If RBCCW is lost for over fifteen (15) minutes, Engineering must be notified to do an evaluation of RCP seals within 24 hours, in order to continued power operation.

Justification

A - Correct; With the "B" RBCCW heat exchanger unavailable, it is NOT possible to align the "B" RBCCW Pump to supply Facility 1 RBCCW. Therefore, the Loss Of RBCCW AOP requires an immediate plant trip, which requires a plant page by the SM.
B - Wrong; A plant page is made whenever a large pump is started for personnel protection. However, there is NO requirement to specifically notify the Aux. building PEO to check the affected pump breaker before restarting the pump, even if it tripped on overload.
C - Wrong; The Loss of RBCCW AOP gives guidance on the cross-tying of facilities, but there is NO requirement to notify plant management before doing so. Delaying the recovery of RBCCW would almost guarantee a plant trip.
D -Wrong; The plant is NEVER intentionally tripped by "forcing an interlock" In this case, the Loss of RBCCW AOP states the plant must be tripped, THEN the two applicable RCPs must be secured However, the NRC would have to be notified if the plant is tripped, especially if it was done in the manner stated in the distracter.

References | **Provided**

AOP-2564, R-4, C-2, St. 3.2 (Inst) & 4.1 (Inst & Cont)
One-Line Diagram of RBCCW System, R-1

Comments and Question Modification History

Modified Distractor 'D' slightly to include tripping the reactor. 11/11/08

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

Generic KIA **Selected**

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

Number 2.4.30 **RO** 2.7 **SRO** 4.1 **CFR Link** (CFR: 43.5 145.11)

Knowledge of events related to system operationlstatus that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

Question #: 9 | **Question ID:** 8000008 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

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

The following initial plant conditions exist:

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

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

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

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

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

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

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

Justification

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

References

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

Comments and Question Modification History

Reworded Distractor **'C'** slightly to state that 'RCS pressure lowering to 2200 psia would cause the backup heaters to reenergize. Provides clarity to distractor.
Reworded Answer 'D' to state that pressurizer spray bypass flow would cause RCS pressure to lower. Provides clarity and makes the correct answer 'more correct.' 11111/08

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

Number AA1.O1 **RO** 4.0 **SRO** 3.9 **CFR Link** (CFR 41.7 145.5 / 45 6)

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

Question #: **10** | **Question ID: 8000009** | ☑ **RO** | ☐ **SRO** | ☑ **Student Handout?** | ☑ **Lower Order?**

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

The plant is at 100% power, steady state, when a grid disturbance causes the main turbine to trip. Before the RO can trip the reactor, he notices that all CEAs are inserting.

Which one of the following, by itself, would indicate that the reactor was shutdown by the ATWS Mitigation Circuit (i.e.; Diverse SCRAM System)?

☐ **A** All eight Trip Circuit Breakers are open.

☐ **B** Both MG Set output contactors are open.

☐ **C** Auto Aux. Feed time delay alarm is active.

☐ **D** One control channel NI is reading >20%.

Justification

B - Correct; The DSS actuation trips both MG set output contactors as an additional way to shutdown the reactor, separate from RPS.
A - Wrong; The TCBs are tripped open, normally, by the RPS. NOT the DSS. This would be "normal" indication that'the reactor tripped.
C - Wrong; The AFAS alarm could be in due to the level shrink on a plant trip driven by a turbine trip. The load reject would cause a spike in SG pressure and result in a higher than expected shrink in SG level.
D - Wrong; Control Channel NIs are an input to the AMSAC, NOT the Diverse Scram System. If control channel NIs are reading >20% in combination with a high pressurizer pressure, then auto aux feed will be initiated.

References **Provided**

LP CED-01-C, R-4, Pg. 7, 9, & 36 (excerpts).

Comments and Question Modification History

Changed distractor 'D' from: Indication the PORVs opened and closed, to: One control channel NI is reading 50%. This is to ensure Distractor 'D' is absolutely incorrect, but plausible due the input to the Automatic Auxiliary Fedwater Actuation vs. the Diverse Scram System.

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

Number EK2.06 **RO** 2.9* **SRO** 3.1* **CFR Link** (CFR 41.7 145.7)

Knowledge of the interrelations between the and the following an ATWS: Breakers, relays, and disconnects

Question #: **11** | ***Question ID:*** ***8054019*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ Student Handout? | 7 Lower Order?

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

The following plant conditions exist:
- A SGTR occurred with all systems operating normally on the plant trip.
- EOP 2525 has been completed and EOP 2534 is being properly implemented.
- The RCS cooldown to less than 515°F has just been completed.

What differences in plant conditions would be seen if the cooldown to less than 515°F had been accomplished with a loss of the RSST at the time of the trip?

☑ **A** SG pressure will be lower in both of the SGs because of the larger RCS Delta-T required when cooling down using natural circulation.

☐ **B** SG pressure will be lower in the ruptured SG because, after the MSIVs are closed, the ruptured SG will be depressurized by the RCS leakage.

☐ **C** SG pressure will be the same, regardless of the type of RCS flow, because the cooldown is always to the same RCS temperature (515°F), with the same amount of decay heat.

☐ **D** SG pressure will be higher in the ruptured SG because of the larger RCS Delta-T required when cooling down using natural circulation.

Justification

A - Correct; The large Delta-T results in a lower TC. SG pressure most closely tracks TC.
B -Wrong; 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 if the SG level is maintained >/= 40% per the EOP-2534.
C - Wrong; It is RCS Tcold that decides SG pressure, NOT how quickly Tcold is dropped or the amount of steaming necessary to drop it.
D - Wrong; The two loops should remain coupled as both SGs are being used in 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.

References

PT E34-01-S, Pg. 12

Comments and Question Modification History

Reworded Distractor 'B' from "...the ruptured SG because the SGs are no longer linked by the Main Steam header and the ruptured SG will be..." to "...the ruptured SG because, after the MSIVs are closed, the ruptured SG will be..." This is to provide clarity to the distractor and still maintain plausibility. 11111/08

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

Number EK1.03 RO 3.9 SRO 4.2 CFR Link (CFR 41 8 / 41.10 145.3)

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

Question #: **12** **Question ID: 8073999** ☑ RO ☐ SRO ☐ Student Handout? ☐ Lower Order?

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

A startup is in progress with the plant near End of Life (EOL), in MODE 1 at 6% power, when a lightning strike in the switchyard causes an automatic plant trip.

During the performance of EOP 2525, the Unit Supervisor (US) receives the following input:

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

Which one of the following administrative limits is in place to prevent this event from exceeding design basis limits?

A Shutdown Margin of greater than or equal to 3.6% delta WK with Tavg greater than 200°F.

☐ **B** Maximum Linear Heat Rate of 15.1 kw/ft while operating in MODE 1.

☐ **C** Minimum average coolant temperature of 515°F while operating in MODES 1 and 2.

☐ **D** Maximum cold leg temperature of 549°F when operating in MODE 1

Justification

A - correct; Shutdown Margin requirements vary throughout core life. The most restrictive condition occurs at EOL, with Tavg at no load operating temperature, and is associated with a postulated Steam Line Break accident and resulting uncontrolled RCS cooldown.
B -wrong; the Linear Heat Rate limit is based on ensuring the peak fuel clad temperature will not exceed 2000°F in the event of a LOCA..
C - wrong; The minimum reactor coolant temperature is based on beginning of life conditions when MTC is slightly positive at low power conditions. This is NOT a consideration at EOL.
D -wrong; the maximum cold leg temperature ensures that the assumed margins to DNB are maintained. In a Steam Line break event, DNB limits are not challenged.

References

Millstone Unit 2 Technical Specifications Bases for SDM. (NOT provided.)
Millstone Unit 2 COLR, SDM Required Value.

Comments and Question Modification History

Replaced Distractor **'C'.** Original was not plausible. 11111/08

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

Generic KIA Selected

NRC KIA Generic **System** 2.2 Equipment Control

Number 2.2.22 **RO** 4.0 **SRO 4.7** **CFR Link** (CFR: 41.5 143.2 / 45.2)

Knowledge of limiting conditions for operations and safety limits.

Question #: **13** | **Question ID:** 8071926 | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

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

Buses 25A and 25B are deenergized due to a failure to automatically fast transfer
Bus 24E is aligned to Bus 24C.
Bus 24C is deenergized; the associated DIG will NOT start (PEO dispatched).
"B" Aux Feedwater pump breaker tripped on fault. (PEO dispatched)
The Terry Turbine tripped on overspeed and will NOT reset. (PEO dispatched)
The Condensate System is NOT in operation.
#2 S/G level is 150 inches and lowering.
#1 SIG level is at 110 inches and lowering.
Trending indicates #1 SG level will be at 70 inches within the next 10 minutes.
All other conditions are as expected.

Early implementation of Once-Through -Cooling __________________

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

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

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

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

Justification

D is correct; Note prior to step 5 of EOP 2537 states:
Once through cooling should be initiated prior to SG wide range level reaching 70 inches if any of the following exists:

1. Main or Auxiliary Feedwater is NOT expected to be restored.
2. Less than two trains of HPSI, PORVs, or ADVs are available.
3. NO Charging Pumps are available.

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

References

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

Comments and Question Modification History

Chanaed #2 SG level from 235 inches to 150 inches and #1 SG level from 150 inches to 110 inches. This is to ensure the examinee realizes that 'early initiation' should begin now and not wait to see if components can be restored before 'early initiation' is attempted.
11111108

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

Number **EA2.2** **RO** 3.0 **SRO** 4.2 **CFR Link** (CFR: 43.5 / 45.13)

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

Question #: **14** | **Question ID: 8080324** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

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

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

A To continue adequate RCS Heat Removal.

☐ **B** To ensure Reactivity Control is maintained.

☐ **C** To restore DC Control Power to components.

☑ **D** To avoid fully discharging the Vital Batteries.

Justification

D is correct; EOP 2530, Step 12, states, " If either vital battery charger is not expected to be restored within one hour, reduce loads on the associated vital battery bus." The station batteries can supply power for a limited time prior to becoming fully discharged. As a result, specific DC loads are secured one hour after the event to allow a more efficient use of the batteries.
A is incorrect; Assumptions used in this event ensure the core will remain covered and cooled for up to 8 hours; therefore, heat removal is not a concern during the first hour.
B is incorrect; The reactor is assumed to be shut down during a station blackout event, therefore, Reactivity Control will be maintained for at least the first hour.
C is incorrect; DC control power is NOT assumed to be lost during a station blackout; therefore, this is not a concern.

References

EOP 2530, Step 12,

N O 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 EA1.05 **RO** 3.3 **SRO** 3.6 **CFR Link** (CFR 41 7 145.5 145.6)

Ability to operate and monitor the following as they apply to a Station Blackout: Battery, when approaching fully discharged

Question #: **15** | **Question ID: 8000003** | ☑ **RO** | ☐ **SRO** | ☐ Student Handout? | ☑ Lower Order?

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

During a Loss of Normal Power (LNP) with a concurrent Safety Injection Actuation Signal (SIAS), various components start on Sequence 3.

During an LNP WITHOUT a concurrent SIAS, NO components start on Sequence 3.

Why is Sequence 3 different for the two conditions?

☐ **A** The load limit for the EDGs is lower for ONLY an LNP than it is for an LNP concurrent with a SIAS, due to SIAS realignment of Service Water.

☐ **B** Starting these components during an LNP WITHOUT a concurrent SIAS could potentially result in an inadvertent, unplanned radioactive discharge to the environment.

☐ **C** The additional components that start on an LNP concurrent with a SIAS are required to mitigate the consequences of a LOCA or ESD event.

☐ **D** The components started in previous sequences for an LNP ONLY situation are larger than the components started in previous sequences for an LNP concurrent with a SIAS.

Justification

C is correct. The components started on Sequence 3 on an LNP with a concurrent SIAS are: LPSI Pump, Containment Spray Pump, and Enclosure Building Filtration Fan. These components are NOT needed for accident mitigation during an LNP ONLY situation.
A is incorrect. Although the Service Water System is realigned during a SIAS to eliminate the heat generated from non-safety related components, the EDG load limit for an LNP ONLY and an LNP with a concurrent SIAS are the same.
B is incorrect. Starting the Enclosure Building Filter Fan on an LNP ONLY would NOT result in an inadvertent unplanned radioactive discharge.
D is incorrect. The electrical loads from previous sequences are actually larger (HPSI Pump) during an LNP with a concurrent SIAS than during an LNP ONLY situation (NO HPSI Pump).

References

LP ESA-01-C, Pg 31
LP ESA-01-C, EDG Load Sequence - No SIAS
LP ESA-01-C, EDG Load Sequence -With SIAS

NO Comments or Question Modification History at this **time.**

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

Number AK3.01 RO 3.5 SRO 3.9 CFR Link (CFR 41.5,41.10 145.6 / 45.13)

Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: Order and time to initiation of power for the load sequencer

Question #: **16** **Question ID: 8000005** ☑ **RO** ☐ **SRO** ☐ Student Handout? ☐ Lower Order?

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

The plant is operating at 100% power with normal conditions when Vital Instrument AC Bus VA-30 is deenergized.

Assuming all equipment responded as designed, what action must the Balance of Plant (BOP) operator take with regard to control of Steam Generator level?

☐ **A** Place " A Main Feed Pump speed control in MANUAL and raise speed.

☐ **B** Continue to monitor " A Main Feed Pump speed and Steam Generator level.

☐ **C** Place BOTH Main Feed Pump speed controls in MANUAL and adjust speeds.

☐ **D** After the plant is manually tripped, start two Auxiliary Feed Pumps.

Justification

B is correct. Although VA-30 is considered a "Vital" control power supply, it is the backup control power supply to the " A MFP speed control. VR-11 is the 'main' power supply and it will continue to supply power to the feed pump speed control, which results in NO change to feed pump speed. The " A MFP Trouble alarm will annunciate when VA-30 is lost, which requires the operator to monitor feed pump speed and SIG level.
A is incorrect. This would be true if VR-11 was not maintaining power to the control system. However, because the control circuitry is NOT impacted by a loss of VA-30, there is no need to place the "A" MFP speed control in MANUAL.
C is incorrect. This is the required SGFP action for a failed steam flow transmitter, which is NOT powered by VA-30, but VR-11/21. Therefore, there is no need to place both feed pump controls in MANUAL. The loss of VA-30 will NOT impact either Feed Pump.
D is incorrect. If VR-11 were also lost, the " A MFP would trip on loss of control power. However, the feed pumps are NOT affected, therefore; the plant, will NOT be tripped.

References

LP MFW-01-C, Pg 37, Table of Power Supplies
ARP-2590D-001, SGFP A Trouble alarm

Comments and Question Modification History

Fixed typo in Distractor 'C'. Changed 'Main Fed Pump' to 'Main Feed Pump'. 11111/08

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

Number AA1.03 RO 3.6* SRO 3.6 CFR Link (CFR 41.7 145.5 145.6)

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

Question #: 17 | **Question ID:** 8000007 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

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

The plant is operating at 100% power in a normal configuration. The AC supply breaker for the "B" Battery Charger suddenly opens. Several minutes later, the "125 VDC Battery 201B Undervoltage" annunciator alarms due to lowering voltage.

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

☐ **A** The output from Inverters 2 and 4 is degraded and causes connected instrumentation to decalibrate.

☐ **B** Control power to Facility 2 AC breakers is lost which results in a loss of breaker protective trip functions.

☑ **C** Battery 201B continues to provide power to DC loads for up to 8 hours after the loss of the Charger.

☐ **D** DC loads powered from Battery 201B begin tripping on undervoltage due to the low voltage condition.

Justification

C is correct. The undervoltage alarm is a annunciated to alert the operator to a degrading condition. The alarm is NOT meant to provide a warning of malfunctioning equipment. Therefore, DC loads are NOT impacted when this alarm annunciates. The Station Batteries are analyzed to provide DC voltage to required loads for at least 8 hours following the loss of its associated charger.
A is incorrect. The output from the inverters is NOT impacted by the lower DC voltage; therefore, connected instrumentation is NOT affected.
B is incorrect. Control power is NOT lost to AC breakers until the battery is depleted. 126 VDC is more than adequate to continue providing control power to the affected AC breakers.
D is incorrect. DC loads powered form Battery 201B will NOT start tripping on undervoltage because DC breakers do not have an undervoltage trip.

References

LP LVD-01-C, Pg 12, Design Basis of Batteries
APR-2590F, 125 VDC Battery 201B Undervoltage alarm

Comments and Question Modification History

Slight change to stem. Changed 'Ten minutes...' to 'Several minutes...'. Ten minutes may be considered too soon for the alarm, which may cause the examinee to think the problem is much more severe.
Changed '...at an indicated 126 volts.' to '...due to the lowering votlage.' The actual alarm setpopint may be confusing to the examinee if helshe believes that the alarm setpoint is different. (NOT required to be memorized.)
In Distractor 'D', changed 'overvcurrent' to 'undervoltage'. Makes the distractor more plausible. Examinee is more inclined to believe components trip on undervoltage than overcurrent. 11111/08

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

Number AA2.02 RO 3.3' **SRO** 3.6 CFR Link (CFR: 43.5 145.13)

Ability to determine and interpret the following as they apply to the Loss of DC Power: 125V dc bus voltage, lowlcritical low, alarm

Question #: **18** | ***Question ID:*** ***79976*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ Student Handout? | ☑ Lower Order?

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

The plant is in normal operation at 100% power. The "LOAD LIMIT LIMITING" light is energized on the C06/7 EHC insert.

Which of the following describes the response of the turbine controls to changes in grid frequency.

☐ **A** The turbine will respond only to a significantly low grid frequency; the turbine speed control unit demand will cause the control valves to open.

☐ **B** The turbine will NOT respond to grid frequency changes in this mode; load limit prevents the turbine speed control unit demand from opening or closing the control valves.

☐ **C** The turbine continuously responds to grid frequency changes in this mode, the control valves open and close in response to speed control unit demand to maintain 60 Hz.

☐ **D** The turbine will respond to a significantly high grid frequency, the turbine speed control unit demand will cause the control valves to close.

Justification

D is correct. The load reference signal and the speed error signal are combined in such a way that even with a "valves wide open" reference signal (+5VDC), the control valves (CV) will close proportionately as speed increases from 100% (60 Hz) to 105% (63 Hz).
A is incorrect. The control valves will NOT open due to a change in load; however, the control valves are able to automatically close proportionally to a load change.
B is incorrect. The control valves WILL respond to an increase in grid frequency by closing proportionally.
C is incorrect. The control valves will NOT open due to a change in load; however, the control valves are able to automatically close proportionally to a load change.

References

LP MTC-01-C, Pg 40 & 42, Main Turbine. Controls on "Load Limit"

NO Comments or Question Modification **History** at this time.

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

Number AK2.07 RO 3.6 SRO 3.7 CFR Link (CFR: 41.4, 41.5, 41.7, 41.10 145.8)

Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Turbine / generator control.

Question #: **19** | ***Question ID:*** ***2100001*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ **Student Handout?** | ☑ **Lower Order?**
Rev. **1** | ☑ **Selected for Exam** | ***Origin:*** **Bank** | ☐ **Past NRC Exam?**

The plant is operating at 100% power and the monthly CEA operability surveillance is in progress. The Reactor Operator (RO) has just finished inserting CEA #1 (Group 7) six steps from the fully withdrawn position, when it suddenly slips to the 126 step position.

Which one of the following combinations of CEAPDS and PPC position indications matches those that would be displayed on C04 under these conditions?

☐ **A** CEAPDS indicates 126 steps Computer indicates 126 steps

☐ **B** CEAPDS indicates 126 steps Computer indicates 174 steps

C CEAPDS indicates 174 steps Computer indicates 126 steps

☐ **D** CEAPDS indicates 174 steps Computer indicates 174 steps

Justification

B - Correct; CEAPDS will display the slipped CEA position because it monitors the reed switches for the individual CEA. However, the PPC will only display a change in CEA position if the CEDM was actually "pulsed" to move the CEA.
A - Wrong; The PPC will not sense the CEA has moved because the CEA slipped to 126 steps and the CEDM was not pulsed to move it.
C -Wrong; This choice has the two indications reversed, if the concept is partially understood.
D - Wrong; This choice assumes both indications need to see the CEDM pulsed to move the CEA.

References

AOP-2556, Pg. 4, Discussion on CEA Pulse Counting possible error.
LP CED-01-C, Pg. 26, explanation of Pulse and Reed CEA position display.

NO Comments or Question Modification History at this time.

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

Number AK2.03 **RO** 3.1* **SRO** 3.2* **CFR Link** (CFR 41.7 145.7)

Knowledge of the interrelations between the Dropped Control Rod and the following: Metroscope

Question #: **20** | **Question ID: 8600020** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

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

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

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

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

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

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

☐ **D** The Level 1 and Level 2 Bistables will reset on Channel " A .

Justification

D - CORRECT: The Level 1/2 bistables will RESET on this channel when the signal drops below 1 X 10-4%. The channel would now be reading 1 X 10-6% power.
A - WRONG: When a Wide Range NI channel drops below 1 X 10-4%, Level 1/2 bistables will ALLOW arming of the Zero Power Mode Bypass, but a bypass key on the channel (not normally in place) must be turned to the "bypass" position. This key would only be in place if testing were being done on the channel, and never during a reactor startup.
B **-** WRONG: The PTTI interlock would indeed be armed for this channel, IF the failed detector power supply were on a Linear Channel.
C - WRONG: The PDIL bypass would activate on this failure, but ALL four channels must activate for the applicable interlocks to be affected.

References

NIS-01-C, Rev. 3, Ch. 2, Pg. 15 of 57, Pg 31 of 57
ARP 2590C-092, Rev. 000

N O Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 032 Loss of Source Range Nuclear Instrumentation

Number AA2.01 **RO** 2.6 **SRO** 2.9* **CFR** Link (CFR: 43.5 145.13)

Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation: Normallabnormal power supply operation

Question #: **21** **Question ID: 8055940** ☑ **RO** ☐ **SRO** ☐ Student Handout? ☐ Lower Order?

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

An Operator is at the Aerated Waste Panel, about to start a radwaste discharge. All applicable administrative requirements have been properly completed and verified up to this point. The Operator starts the sample pump and the "AERATED WASTE EFFLUENT RADIATION HI" annunciator clears, but the "RM9116 LOSS OF FLOW" annunciator clears and then immediately re-alarms.

Then, the Operator turns the control switch for the first discharge valve directly to OPEN but the red light does NOT light. The Operator then turns the hand switch for both discharge valves in the CLOSE direction first, then to OPEN. The red lights on both valves energize and, a couple seconds later, the green lights on both valves go out.

The Operator continues the discharge with the following observations:

- Flow indication on the discharge flow recorder is about half what was seen on the last discharge.
- Aerated Waste Monitor Tank level is lowering at a rate expected for the lower discharge flow.
- The Aerated Rad. Waste Discharge filter delta-P is much higher than that seen on the last discharge.
- All recorded parameters appear to be within acceptable ranges and tracking normally.

What is the status of the Aerated Waste Discharge Radiation Monitor and why?

☐ **A** Operable; the abnormally low discharge flow will NOT adversely affect the radiation monitor sample flow.

☐ **B** Operable; the abnormally low discharge flow will result in a more conservative radiation monitor reading.

☐ **C** Inoperable; the discharge will isolate ONLY on a loss of control power to the discharge valves.

☐ **D** Inoperable; the discharge will isolate ONLY on a loss of power to, or failure of, the radiation monitor.

Justification

C - Correct; Starting the sample pump often clears the low sample flow alarm but then triggers the high sample flow condition as sample flow stabilizes. Although this sample flow fluctuation would normally isolate the discharge, over-riding a high sample flow alarm also prevents a low sample flow condition from closing the valves. The sample flow and radiation alarms on PIOPS must be acknowledged and cleared BEFORE the discharge valves are opened, or opening them means they have been "over-ridden" open. Based on the conditions stated, the discharge valves HAVE BEEN over-ridden open, and will NOT close for ANY alarm condition triggered by the Rad. Monitor.
A - Wrong; Because the discharge valves were overridden, they will NOT close due to a high radiation monitor reading; therefore, the Rad Monitor is inoperable. The rad. Monitor sample flow is a separate slip stream driven by a sample pump. The sample flow would NOT be effected by the discharge flow rate.
B - Wrong; Because the discharge valves were overridden, they will NOT close due to a high radiation monitor reading; therefore, the Rad Monitor is inoperable. The low discharge flow may result in a greater sampling of the rad. waste as it passes by the sample pump suction. However, this potential "over-sampling" could only result in an artificially high radiation reading, which is more conservative.
D - Wrong; The discharge valve over-ride is designed to allow for a rad. waste discharge with the rad. monitor de-energized Therefore, a radiation monitor failure of any kind will NOT secure the discharge.

References

RMS-00-C, Radiation Monitoring, Rev. 7, Ch. 2
ARP 2593A, Rev 1, Ch. 3
ALR-04-C, Aerated Liquid Radwaste, Rev. 3, Ch. 1

Comments and Question Modification History

Reworded the stem to eliminate the second operator. Question now requires the examinee to determine whether or not the rad monitor is operable or not. 11/11/08

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

Number AK3.03 RO 3.0 SRO 3.7 CFR Link (CFR 41.5,41.10 145.6 145.13)

Knowledge of the reasons for the following responses as they apply to the Accidental Liquid Radwaste Release: Declaration that a radioactive-liquid monitor is inoperable

Question #: **22** | ***Question ID:*** ***8000055*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

A control room evacuation is required due to toxic gas. All required immediate actions of AOP 2551, Shutdown from Outside the Control Room, have been performed and the operating crew has gathered at C-21, Hot Shutdown Panel.

It is subsequently determined that a cooldown from outside the Control room is necessary, which will require Manual Boration to ensure Shutdown Margin is maintained.

What is the reason for opening the Gravity Feed Valves, CH-508 and 509, locally in manual?

☐ **A** These valves were deenergized after the Control Room was evacuated.

☑ **B** There are NO controls for these valves on Hot shutdown Panel, C-21.

☐ **C** The valves for the Boric Acid Pumps CANNOT be locally aligned to the Charging Pumps.

☐ **D** These valves can be easily controlled to provide precise control of Boric Acid flow rate.

Justification

B is correct The Gravity Feed Valves can only be operated from C-02 and will automatically open on a SIAS. There are NO controls on C-21 for the Gravity Feed Valves.
A is incorrect. The Gravity Feed Valves are NOT deenergize after the Control Room evacuation. If The Control Room evacuation were to occur due to an Appendix "R" fire, then the valves would be deenergized.
C is incorrect. The valves associated with the Boric Acid Pumps can be manually aligned locally; however, there are NO Boric Acid Pump controls on C-21.
D is incorrect. The Charging Pumps determine the flow rate for Boric Acid. Throttling these valves will NOT provide any flow rate control.

References

AOP-2551, Pg. 16, Step 1.6c

(NO Comments or Question Modification History at this

NRC K/A System/E/A **System** 068 Control Room Evacuation

Number AK3.01 **RO 3.9** **SRO** 4.2 **CFR Link** (CFR 41.5,41.10 145.6 145.13)

Knowledge of the reasons for the following responses as they apply to the Control Room Evacuation: System response to reactor trip

Question #: **23** | ***Question ID:*** ***8000061*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ student Handout? | ☐ Lower Order?

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

The plant has just started up from a refueling outage and is stable at 30% power on a secondary chemistry hold.

Then, DC bus 201B de-energizes due to a bus fault, resulting in the following conditions:

- Both MSIVs close
- The "B" main Steam header ruptures in containment
- 24B and 24D are de-energized (along with all lower voltage busses powered by them)
- Facility One SIAS, CIAS, EBFAS, MSI and CSAS have all fully actuated
- All other plant systems and components that have power are functioning as designed.

The crew is evaluating numerous alarms and indications caused by the power loss and subsequent ESD.

Which of the following alarm indications will require contingency actions be taken to prevent exceeding a design limit?

☐ **A** C05 alarms indicating an ESD on #2 SG and C08 alarm indicating VR-21 is de-energized.

☐ **B** C02/3 alarms indicating RCS Th and Tc are abnormally low and the BASTs gravity feed valves are de-energized closed.

☐ **C** C04 alarms indicating Facility One Aux. Feedwater has actuated and C08 alarm indicating loss of DV-20.

☐ **D** C01 alarms indicating CTMT Spray has actuated and C01 indicating only two CAR fans and one CS pump are operating.

Justification

C - CORRECT; All alarms and indications mentioned in the four choices are expected for the given event, a loss of DC bus 201B and subsequent ESD on the " B Main Steam header. However, Choice "C" information indicates Auxiliary Feedwater will be feed the affected steam generator. The Design Basis ESD in CTMT states that ALL feed to the affected steam generator must be secured within 30 minutes to meet the design criteria for CTMT Integrity. In this criteria, only one facility of ESAS equipment is assumed to be functioning and available.
A - WRONG; VR-21 is deenergized, based on the given event. However, this would prevent the "B" Atmospheric Dump Valve (ADV) from being operated from the control room. If the other steam header was ruptured, this would be the correct choice, as it would require immediate action to get an operator to C21 (Remote Shutdown Panel) to control RCS temperature when the affected SG boils dry (thus preventing PTS)
B - WRONG; This gives indication of an excessive cooldown of the RCS with a potential problem with boric acid injection. However, the other facility of power is available to allow automatic alignment of a boric acid source to the remaining charging pump, which is sufficient (although not optimum) to meet "reactivity control". Procedure steps will ensure additional boron injection is aligned, but this is above the required amount.
D - WRONG; One facility of CTMT Cooling and Pressure Control is certainly NOT optimum during and ESD, but it is designed to be sufficient to maintain CTMT Integrity, provided all feed is secured to the affected SG in the required time frame.

References

OP-2260, Pg. 43; ESD Mitigation Requirements and Critical Tasks

NO Comments or Question Modification History at this **time.**

NRC K/A System/E/A System 069 Loss of Containment Integrity

Generic KIA **Selected**

NRC K/A Generic System Emergency Procedures /Plan

Number 2.4.45 RO 3.3 SRO 3.6 CFR Link (CFR: 43.5 145.3 / 45.12)

Ability to prioritize and interpret the significance of each annunciator or alarm.

Question #: **24** | ***Question ID:*** *8000056* | ☑ ***RO*** | ☐ ***SRO*** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The plant was at 100% power when a leak developed on the charging header connection to RCS loop one. The crew attempted to isolate CVCS from the leak, but charging header isolation to loop one, CH-519, would NOT close. All other charging header valves were successfully closed, but The leak subsequently degraded to a rupture, resulting in a Small-Break LOCA.

The following conditions now exist:

- RCS pressure is 1400 and lowering slowly.
- The crew has transitioned to EOP-2532, Loss Of Coolant Accident.
- The US has determined that safety injection flow is NOT adequate.
- All other plant equipment is operating as designed for the present plant conditions.

Which of the following alignments would raise safety injection flow?

☑ **A** Align the charging pumps discharge to the " A HPSI injection line.

☐ **B** Align the " A HPSl pump to inject through the auxiliary spray line.

☐ **C** Align the charging pumps to inject through the auxiliary spray line.

☐ **D** Align the charging pumps to discharge to the RCS loop two header

Justification

A - CORRECT; SB-LOCA, has the potential to cause RCS Inventory to be lost faster than RCS pressure. With RCS pressure > 1250 psia (HPSI shutoff head), the charging pumps are the only pumps capable of injecting into the RCS. Based on the leak location and the failure of CH-519 to close, all three normal charging header injection paths are lost. Therefore, the alternate injection path, through " A HPSI injection line, must be used.
B - WRONG; This path could be used if RCS pressure were lower and the leak were in a different location.
C - WRONG; This path would lower RCS pressure and allow HPSI to inject, but the leak location and CH-519 failure prevents it.
D -WRONG; The RCS loop injection lines are separate, and protected by check valves and isolation valves, but the leak location and failure of CH-519 eliminate this option.

References

CVCS One-Line Diagram
OEP-2541, Pre-SRAS SI Flow Curve

N O Comments or Question Modification History at this time.

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

Number EA1.09 **RO** 3.7 **SRO** 3.8 **CFR Link** (CFR 41.7 145.5 145.6)

Ability to operate and monitor the following as they apply to a Inadequate Core Cooling: CVCS

Question #: **25** | ***Question ID:*** ***8000012*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The plant is at 100% power, steady state, when a normally scheduled primary sample shows high fission product activity in the RCS.

Chemistry department has recommended raising charging and letdown flow to the maximum limit to clean up the RCS.

Which of the following describes a required action and why that action must be accomplished?

☐ **A** The second letdown flow control valve must be placed in service in order to raise letdown flow to the maximum.

☑ **B** Health Physics department must be notified of changes in letdown flow because this will change area radiation levels.

☐ **C** Chemistry must verify RCS boron concentration within six hours due to a potential change from raising letdown flow.

☐ **D** A second ion exchanger must be placed in service during maximum letdown flow to limit the ion exchanger delta-P.

Justification

B - Correct; Changing letdown flow will change radiation levels in the -5' penetration area because that is were the letdown line first comes out of CTMT. This is a procedure required ALARA concern, especially important when the RCS is known to be at **a** higher activity.
A - Wrong; The second letdown valve is only placed in service if low RCS pressure precludes raising letdown flow to the desired amount. There is NO event given that would lower RCS pressure to this level.
C -Wrong; There is a requirement in OP-2204 that when power is going to be changed by **>/=** 15% in one hour, the RCS must be sampled for IODINE within **2 -** 6 hours to check for potential fuel pin leakage. This procedure also directs that letdown flow be increased to allow for a smooth power change, but the two requirements are not connected.
D **-** Wrong; The IXs are in series, therefore a second IX in service would not effect the delta-P across a single exchanger.

References

AOP-2511, Pg. 5, HP notify requirement for letdown flow changes.

NO Comments or Question Modification History at this time.

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

Number AK3.05 **RO** 2.9 **SRO** 3.6 **CFR Link** (CFR 41.5,41.10 145.6 145.13)

Knowledge of the reasons for the following responses as they apply to the High Reactor Coolant Activity: Corrective actions as a result of high fission-product radioactivity level in the RCS

Question #: **26** | ***Question ID:*** ***8680010*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The plant was manually tripped from 100% power due to a 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 (except for ALL loads on VA-20, which are still deenergized).

The crew has transitioned to EOP-2536, Excess Steam Demand Event, and has carried out all applicable steps.

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.

Justification

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 (CTMT).
A - Wrong; 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.
C - Wrong; The #1 ADV control power has been modified to allow operation in manual only, from the C05 controller upon a loss of VR-11. However, the #2 ADV C05 controller is powered from VA-20 and would be de-energized.
D - Wrong; The C10 Fire Shutdown panel is designed for used when the control room must be evacuated due to an during 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 here.

References

AOP-2504D (Loss of VA-20), Pg. 3, Discussion of #2 ADV loss of control.

NO Comments or Question Modification History at this time.

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

Number AK1.l **RO 3.1** **SRO 3.3** **CFR Link** (CFR: 41.8 141.10 145.3)

Knowledge of the operational implications of components, capacity and function of emergency systems as they apply to the RCS Overcooling.

Question #: 27 **Question ID: 876163** ☑ RO ☐ SRO ☐ Student Handout? ☐ Lower Order?

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

The plant was tripped due to an Excess Steam Demand event inside the turbine building. During the performance of EOP 2525, the following additional conditions were noted:

* Three (3) CEAs stuck fully withdrawn.
* Reactor power is 5 X 10-2% and stable.
* The normal charging flow path is NOT available due to a pipe rupture between CH-429, Charging Header Isolation Valve, and the containment wall.
* Pressurizer pressure = 1750 psia and rising.
* Pressurizer level = 26% and rising.
* RCS Tavg = 490 °F and rising.

Which of the following subsequent procedures is required to be used for guidance to meet the highest priority safety function?

☐ **A** AOP-2558; Emergency Boration

☐ **B** EOP 2541, Appendix 3; Emergency Boration

☐ **C** AOP-2512; Loss Of All Charging

☐ **D** EOP-2540A; Functional Recovery of Reactivity

Justification

D - Correct; EOP 2540A has guidance for use of the alternate charging flow path through the safety injection header. Upon completing EOP-2525, the crew must transition to an event driven EOP for subsequent guidance on a mitigation strategy. The stated conditions require the use of the alternate charging header, which is addressed in EOP-2540A.
A - Wrong, AOP-2558 does provide guidance for boric acid injection using the alternate charging flow path through the SI header. However, procedure usage guidelines require the completion of EOP-2525 before an AOP is referenced for guidance in event mitigation.
B - Wrong; EOP-2541, Standard Appendices, provides guidance for the boric acid injection in EOP space. However, the normal charging header must be available.
C - Wrong; Although "all charging" has been lost, AOP-2512 is NOT written to recover charging flow within an EOP event. The assumption with the AOP is that the loss of charging flow is the event of concern.

References

EOP-2541, App. 1, Diagnostic Flow Chart for No Reactivity Control.
EOP-2540A, Pg. 6, St. 2, Charging Pump alignment to alternate path.

(NOComments or Question Modification History at this time.

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

Number EA2.1 RO 3.2 SRO 4.4 CFR Link (CFR: 43.5 / 45.13)

Ability to determine and interpret facility conditions and selection of appropriate procedures during abnormal and emergency operations as they apply to the Functional Recovery.

Question #: **28** | ***Question ID:*** ***78320*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ Student Handout? | ☐ **Lower** Order?

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

Which of the following will cause an RCP lift pump to start?

☑ **A** Repositioning the lift pump handswitch from "STOP" to "AUTO" with the associated RCP breaker already open.

☐ **B** Repositioning the lift pump handswitch from "STOP" to "AUTO" with the associated RCP breaker already closed.

☐ **C** Placing the associated RCP handswitch in "START" with the associated lift pump handswitch already in "AUTO" position.

D Placing the associated RCP handswitch in "STOP" with the associated lift pump handswitch already in "STOP" position.

Justification

A is correct. RCP lift pump starts when the lift pump handswitch is in the "AUTO position and the RCP breaker is in the open position. Normal operation has the Lift pump hand switch in AUTO with the RCP running, such that the Lift pump will auto-start when the RCP trips or is turned off.
B is incorrect. The Lift Pump will NOT automatically start by placing the associated handswitch in AUTO from the STOP position with the RCP breaker in the closed (or open) position.
C is incorrect. Starting an RCP with the Lift Pump in AUTO will NOT automatically start the Lift Pump. The Lift Pump must be started 2 minutes prior to starting associated RCP.
D is incorrect. Stopping an RCP with the associated Lift Pump in STOP will NOT automatically start the Lift Pump. The Lift Pump must be in AUTO when stopping the associated RCP.

References

RCS-00-C, RCP Lift Pump, Rev. 8, Ch 1, Pg. 41

NO Comments or Question Modification History at this time.

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

Number A4.03 RO 2.8 SRO 2.5 CFR Link (CFR: 41.7 / 45.5 to 45.8)

Ability to manually operate and/or monitor in the control room: RCP lube oil and lift pump motor controls

Question #: **29** | *Question ID:* 78836 | ☑ *RO* | ☐ *SRO* | ☐ Student Handout? | ☐ Lower Order?

Rev. 0 | 3 Selected for Exam | *Origin:* *Bank* | ☐ Past NRC **Exam?**

The plant is at 100% power, normal operation, when the Letdown Backpressure Controller, PIC-201, transmitter fails to 200 psig.

Which one of the following is the system response to this instrument failure?

☐ **A** Indicated letdown flow will remain constant and letdown backpressure will indicate 600 psig.

☐ **B** Letdown backpressure will indicate 600 psig and Pressurizer level will slowly rise.

☐ **C** Letdown backpressure will indicate 200 psig and letdown flow will go to approximately 140 gpm.

☑ **D** Indicated letdown flow will go to 0 gpm and Pressurizer level will remain constant.

Justification

D is correct; The failure of the backpressure controller transmitter will cause the backpressure control valve to close causing the Letdown line relief valve, 2-CH-345, to lift. Because the relief valve is located upstream of the letdown flow indicator, indicated letdown flow will go to zero. The Letdown relief will still pass the flow allowed by the Letdown flow control valve and Charging flow is NOT affected; therefore, Pressurize level will NOT be affected.
A is incorrect. If the Letdown Backpressure Controller, PIC-201, transmitter fails to a value below the desired pressure of 300 psig, then the backpressure control valve will close and cause actual backpressure to exceed the lift setting of the Letdown Line Relief Valve; however, indicated pressure will read whatever pressure the transmitter has failed to. Because the Letdown flow indicator is downstream of the relief valve, letdown flow will indicate 0 gpm even though actual letdown flow through the relief valve is hasn't changed.
B is incorrect. If the Letdown Backpressure Controller, PIC-201, transmitter fails to a value below the desired pressure of 300 psig, then the backpressure control valve will close and cause actual backpressure to exceed the lift setting of the Letdown Line Relief Valve; however, indicated pressure will read whatever pressure the transmitter has failed to. Because letdown and Charging flows are still matched, Pressurizer level will NOT change.
C is incorrect. If the Letdown pressure transmitter fails to 200 psig, then letdown backpressure will read 200 psig; however, letdown flow will NOT go to 140 gpm regardless of how the backpressure transmitter fails. The Letdown flow control valve will maintain approximately 40 gpm.

References

OP-2304A, Pg. 4, Precaution 3.5
CVC-00-C, Pg

NO Comments or Question Modification **History** at this time.

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

Number K4.11 **RO** 3.1 **SRO** 3.6 CFR Link (CFR: 41.7)

Knowledge of CVCS design feature(s) and/or interlock(s) which provide for the following: Temperaturelpressure control in letdown line: prevent boiling, lifting reliefs, hydraulic shock, piping damage, and burst

Question #: **30** **Question ID:** 8200015 ☑ **RO** ☐ **SRO** ☐ Student Handout? ☐ Lower Order?

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

Initial Conditions:

- The plant has tripped due to a state wide blackout (the grid is lost).
- three (3) CEAs are stuck fully withdrawn.
- 24D is de-energized due to a bus fault.
- 24E is aligned to 24D.
- "B" Charging pump is aligned to Facility 2.
- The RO has successfully initiated emergency boration using the " A charging pump.
- ALL other plant equipment has responded as designed.

If the " A charging pump discharge relief valve were to stick full open at this time, which of the following describes an action needed to allow concentrated boric acid to be injected into the RCS and the reason for that action?

☐ **A** Align charging pump flow to Aux. Spray to reduce the charging pump's discharge back pressure.

☑ **B** The "B" charging pump must be started on Facility 1; " A & "C" charging pumps are unavailable.

☐ **C** " A charging pump must be aligned to the alternate charging path; Facility 2 pumps are unavailable.

D HPSI must be used for boron injection; the discharge relief capacity exceeds three charging pumps.

Justification

B - Correct; The loss of 24D de-energizes the "C" charging pump, and the "B" charging pump due to its initial power supply alignment. Each charging pump has its own discharge relief valve, which when lifting is designed t relieve the entire capacity of the respective pump. As has been seen on MP2, when this valve fails open, 100% of the flow from the applicable charging pump is diverted to Clean Liquid Radioactive Waste. Procedural guidance states that if the "B" charging pump is not available solely because of its present power supply alignment, its must be shifted to power from the other facility and started.
A - Wrong; Reducing RCS pressure is the eventual requirement if there are NO charging pumps available. In that instance, Choice " D may be correct.
C -Wrong; This would be the choice if the "B" charging pump were NOT available for reasons OTHER than loss of power to the facility it was presently aligned to.
D - Wrong; The LETDOWN line relief valve has the capacity to relieve all three charging pumps. However, each charging pump has its own discharge relief valve that only relieves that applicable charging pumps discharge.

References

EOP-2541, App 3, Pg. 3, Emergency Boration w/ "B" CCP.

Comments and Question Modification History

Replaced Choice "A" [Isolate RCP bleedoff flow; the discharge relief ensures pump minimum flow of approximately 4 gpm.] (too easy) 11111108

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

Number A2.14 RO 3.8* SRO 3.9 **CFR** Link (CFR: 41.5/ 4315 14513 14515)

Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Emergency boration

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

Question #: **31** | ***Question ID:*** ***55096*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ Student Handout? | ☐ Lower Order?

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

The plant is on shutdown cooling, with RCS pressure being maintained manually.

Pressurizer (PZR) level is 40%, as seen on the Cold Cal. Indication L-103.
Channel "X" and "Y" PZR level indicate 47% on both controllers.
ALL Pressurizer Backup Heater control switches are in "PULL-TO-LOCK.

Which one of the following actions is required to control PZR pressure using the Proportional Heaters?

☐ **A** The pressure controller setpoint must be lowered to the desired pressure.

☑ **B** The Proportional Heater Breakers must be opened and closed as needed.

☐ **C** The pressure controller output must be MANUALLY adjusted as necessary.

☐ **D** The PZR Heater Selector Switch must be selected to the "X & Y" position.

'Justification

B is correct. With RCS temperature in the ranae for SDC operation, the Reactor Regulating System would calculate a PZR setpoint of 40%. However, because the PZR level control channels are calibrated for NOT/NOP, they would indicate a level of 47%. 47% 40%=7% mismatch in level. The PZR Level Control System will see this mismatch as a "level insurge" and respond accordingly. With level >/= 3.6% above setpoint, the response will cause all Proportional Heaters to come on at maximum output, regardless of the Pressure Controller's output, unless they are manually secured by opening their individual breakers.
A is incorrect. The minimum pressure setpoint available on the pressure controller is 1500 psia, far above any value required at this mode. However, because the "insurge" relay that causes the proportional heaters to be at rnax. output does not receive an input from any level or pressure controller, neither the pressure controller nor the level controller will have any effect on it.
C is incorrect as with the AUTO setpoint, the minimum pressure setpoint available on the pressure controller is 1500 psia; therefore, the proportional heaters cannot be adjusted in manual to maintain pressure low enough for SDC operation.
D is incorrect. Placing or ensuring the PZR Heater Selector Switch is in the "X & Y" position will not provide any control of the proportional heaters. This switch is used only when one of the two control power supplies is unavailable or a specific channel of level control is considered failed.

References

0P-2204, Attachment. 4, PZR Level Control Program

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Corrected t in Γ -'C'. adjust to ׀j 11/11/08

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

Number A4.03 **RO** 2.8* **SRO** 2.7* CFR **Link** (CFR: 41.7 145.5 to 45.8)

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

Question #: **32** | **Question ID: 8000015** | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

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

The plant tripped from 100% power. On the trip, the supply breaker to Bus B-51 tripped and CANNOT be reset. While the crew was performing EOP 2525, Standard Post Trip Actions, a Large Break LOCA developed. All other systems and components operated as expected. The crew subsequently entered the appropriate procedure.

What action is required to ensure maximum obtainable LPSI flow is established?

☐ **A** Dispatch a PEO to open the Facility 1 LPSI Injection Valves.

☐ **B** Override and open all four LPSI Injection valves from C-01.

☐ **C** Manually open the Facility 1 LPSI Injection valves from C-01.

☐ **D** Multiply the Facility 2 LPSI flow indication by a factor of two.

Justification

A is correct. Bus 8-51 supplies power to the Facility 1 LPSI Valves. Unlike the HPSI Injection valves which are maintained open, the valves will automatically open on SIAS. Due to the loss of power, the only way to open ihe valves is in the local, manual mode. All other LPSI components should function as designed to ensure maximum flow is obtained.
6 is incorrect. The Facility 2 LPSI Injection Valves opened on the SIAS. The Facility 1 LPSI Injection Valves will NOT open from C-01 due to the loss of 6-51.
C is incorrect. The Facility 1 LPSI Injection Valves will NOT open from C-01 due to the loss of 6-51.
D is incorrect. The LPSI Facility flow instruments are powered from VR-1land VR-21, respectively. Therefore, the LPSI flow curve gives guidance to double indicated flow to get an accurate indication when either VR-11 or VR-21 are lost. However, indicated LPSI flow is real, because the flow control valves are still closed on the side with deenergized flow indication.

References

LP ECC-01-C (ECCS) , Pg. 12, LPSI Injection Valve discussion

NO Comments or Question Modification History at this time.

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

Number K2.04 **RO** 3.6 **SRO** 3.8 **CFR Link** (CFR: 41.7)

Knowledge of bus power supplies to the following: ESFAS-operated valves

Question #: **33** | ***Question ID:*** ***8054464*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ Student Handout? | ☑ Lower Order?

Rev. **0** | ☐ Selected for Exam | ***Origin:*** ***Mod*** | ☐ 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.

Which of the following is required to ensure the Quench Tank will be maintained within its design limits?

☐ **A** The Quench Tank must be aligned to continuously drain to the PDT to maintain level below the maximum limit.

☑ **B** Quench Tank cooling must be manually initiated, as required, to maintain temperature below the design limit.

☐ **C** The Quench Tank pressure regulator must be aligned to continuously maintain pressure below the design limit.

☐ **D** Quench Tank gas space must be regularly sampled to ensure hydrogen concentration is below the design limit.

Justification

B - 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.
A -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.
C - Wrong; The pressure regulator for the Quench Tank is not designed to function in an automatic mode, and leaving it open could result in the generation of excessive amounts of gaseous rad. waste.
D - Wrong; The Quench Tank gas space is expected to contain a high concentration of hydrogen. As water from an RCS Safety or PORV enters the tank, it will depressurizes and the entrained gasses will come out of solution.

References

OP-2301A (PDT & Quench Tank Operation), Pg. 4, CautionlInstruction of valve control when cooling

Comments and Question Modification History

Fixed typo in stem. Changed "...slowly rise..." to "...slow rise..." 11111108

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

Number A1.03 RO 2.6 SRO 2.7 **CFR** Link (CFR: 41.5145.5)

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:** **79028** | ☑ **RO** | ☐ **SRO** | ☐ Student Handout? | ☐ Lower Order?

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

With the plant operating at 100% power with Bus 24E aligned to Bus 24C, the following alarms are received within a 5 minute period of time:

- RBCCW HDR B PRESS LO (C-0617)
- RBCCW HDR B FLOW HI (C-0617)
- RBCCW SURGE TK LEVEL HI/LO (C-0617)
- AUX BLDG SUMP LEVEL HI (C-0617)
- PMW HEADER LOW PRESSURE (C-0213)
- Various low flow annunciators for components supplied by "B" RBCCW header

NO other annunciators are in alarm.

Which of the following conditions caused these indications and what actions, per AOP 2564, Loss of RBCCW, will be required to mitigate the consequences of this event?

☐ **A** The RBCCW supply piping has ruptured at the'inlet to the "C" RBCCW Heat Exchanger.
Align the "B" RBCCW Pump and Heat Exchanger to supply Facility 2 and place them in service; Isolate "C" RBCCW Heat Exchanger and place "C" RBCCW Pump in Pull-To-Lock.

☐ **B** The RBCCW header has ruptured on the discharge piping that connects directly to the "C" RBCCW Heat Exchanger outlet.
Isolate the "C" RBCCW Heat Exchanger, secure RBCCW Surge Tank make up, 2-RB-215, trip the reactor, secure the "B" and "D" RCPs, and perform Standard Post Trip Action, EOP 2525.

☐ **C** The RBCCW Supply piping from the RBCCW Surge Tank to "B" Header has ruptured.
Close the RBCCW Surge Tank Supply to the "B" Header, open the "C" RBCCW Pump suction from the "A" Supply Header, and close the "C" RBCCW Pump suction from the "B" Supply Header.

☐ **D** The RBCCW piping that supplies the Letdown Heat Exchanger, Sample Coolers, and the Degasifier has ruptured.
Place the "C" RBCCW Pump in Pull-To-Lock, secure RBCCW Surge Tank Make Up, 2-RB-215, trip the reactor, secure the "B" and "D" RCPs, and perform Standard Post Trip Actions, EOP 2525.

Justification

D is correct. The RBCCW low pressure alarm is indicative of a header rupture downstream of the RBCCW Pump. The RBCCW high flow would narrow down the rupture to downstream of the flow instrument which is downstream of the RBCCW heat exchanger. The various low flow annunciators for components supplied by the "B" RBCCW header would further narrow down the location to a component supplied by the "B" RBCCW header.
A is incorrect. A rupture of the "B" RBCCW header at the inlet to the RBCCW heat exchanger would NOT result in a high flow alarm. Additionally, an RBCCW sump high level annunciator would also alarm.
B is incorrect. The RBCCW header flow instrument (and annunciator) are located downstream of the RBCCW heat exchanger discharge isolation valve; therefore, a low flow annunciator would alarm for this condition. Additionally, an RBCCW sump high level annunciator would also alarm.
C is incorrect. A rupture in the suction piping to the RBCCW Pump would NOT result in an RBCCW header high flow alarm.

References

ARP for RBCCW Hi Flow alarm

Comments and Question Modification History

Discussed the need to add Letdown High Temperature alarm to the list of annunciataors. Determined that it was NOT appropriate. Annunciator will NOT alarm for this condition. 11111108

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

Number A2.07 RO 2.5' SRO 2.8' **CFR** Link (CFR: 41.5 143.5 145.3 145.13)

Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Consequences of high or low CCW flow rate and temperature; the flow rate at which the CCW standby pump will start

Question #: **35** | **Question ID: 8066598** | ☑ **RO** | ☐ **SRO** | ☐ Student Handout? | ☑ Lower Order?

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

The plant was operating at 100% power with all equipment in a normal alignment. Bus 24C is supplying Bus 24E. The "C" RBCCW Pump suddenly trips on fault. The Balance of Plant (BOP) operator starts the "B" RBCCW Pump to supply Facility 2 and places the 'C' RBCCW Pump handswitch in Pull-To-Lock. NO other operator action is taken.

Subsequently, a small break LOCA occurs followed by a plant trip and Loss Of Offsite Power (LNP) SIAS, CIAS, EBFAS, MSI and UV actuate and all equipment functions as designed.

Which of the following correctly describes the condition of the RBCCW Pumps and Heat Exchanger TCVs following this event?

☐ **A** The 'A' and 'B' RBCCW Pumps will be in operation with all three heat exchanger TCVs full open.

☐ **B** The 'A' and 'B' RBCCW Pumps will be in operation with only the 'A' and 'C' heat exchanger TCVs full open.

☑ **C** Only the 'A' RBCCW Pump will be in operation with all three heat exchanger TCVs full open.

☐ **D** Only the 'A' RBCCW Pump will be in operation with only the 'A' and 'C' heat exchanger TCVs full open.

Justification

C is correct. A SIAS will result in the start of the 'A' RBCCW Pump. The 'B' RBCCW Pump SIAS/LNP Block handswitch at the breaker was left in BLOCK which will NOT allow the "B" RBCCW Pump to start on SIAS. All three RBCCW heat exchanger TCVs will get an open signal on the SIAS.
A is incorrect. The 'B' RBCCW Pump will NOT start on SIAS due to the SIAS/LNP Block handswitch being left in the BLOCK position.
B is incorrect. The 'B' RBCCW Pump will NOT start on SIAS due to the SIAS/LNP Block handswitch being left in the BLOCK position. Additionally, all three RBCW heat exchanger TCVs will get an open signal on the SIAS.
D is incorrect. While it is true that only the 'A' RBCCW Pump will start on the SIAS, all three RBCW heat exchanger TCVs will get an open signal on the SIAS.

References

AOP-2564, Pg. 15 + 16
ARP-2590-097, "B" RBCCW Pump SIASILNP Manually Blocked

Comments and Question Modification History

Modified stem to state an LNP occurred on the trip from the LOCA

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

Number A3.08 RO 3.6' SRO 3.7* CFR Link (CFR: 41.7145.5)

Ability to monitor automatic operation of the CCWS, including: Automatic actions associated with the CCWS that occur as a result of a safety injection signal

Question #: **36** | ***Question ID:*** ***8000010*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ Student Handout? | ☐ Lower Order?

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

While performing a calibration on the Acoustic Valve Monitoring System (AVMS) on RC-05E, Accident Monitoring Panel, I&C personnel discovered that the system had lost power due to a blown fuse. The operators have taken all the appropriate administrative actions and I&C is continuing to troubleshoot the cause of the blown fuse.

If the plant were now to experience a condition that caused a Pressurizer Safety Valve to open, what indication(s) must the operators use to determine that only a Safety Valve, and NOT a Power Operated Relief Valve, is open?

☐ **A** Quench Tank level, pressure, and temperature would rise.

☐ **B** The associated 'Safety Valve Open" annunciators on C-0213.

☑ **C** The temperature indicators on C-0213 downstream of the safeties.

☐ **D** The red and green position indication lights for the safeties.

Justification

C is correct. The tailpipe temperatures downstream of each safety valve (recent modification) are powered from a different source than the AVMS and would provide adequate indication to determine which safety had opened. The temperature for the open safety should rise to the saturation temperature for the pressure in the Quench Tank.
A is incorrect. Quench Tank level, temperature, and pressure will indicate an input to the tank from the Pressurizer; however, it's impossible to determine which of the four valves opened (2 safeties; 2 PORVs) as all four relieve there.
B is incorrect. The 'Safety Valve Open" annunciators on C-02/3 are triggered by the AVMS.
D is incorrect. There are no red or green lights for the safeties. Only the PORVs have position indication lights. (Recent modification)

References

Picture of C02/3 Vertical, RCS Safety & PORV indications.

Comments and Question Modification History

Added a noun name to RC-05E. 11/11/08

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

Number K2.04 **RO** 2.7* **SRO** 2.9* CFR Link (CFR: 41.7)

Knowledge of bus power supplies to the following: Indicator for code safety position.

Question #: **37** | **Question ID: 1000022** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

I&C is performing a function test on RPS channel 'A' and has bypassed the power related trip on the channel. Vital instrument bus VA-30 is suddenly lost.

Based on the above, what is the resulting condition of the RPS?

☐ **A** The loss of Channel 'C' causes four TCBs to open, but the reactor has NOT tripped.

☐ **B** With Channel 'A' bypassed and Channel 'C' de-energized, ALL TCBs remain closed.

☐ **C** Coincident trip signals are processed from Channels 'A' and 'C' resulting in a reactor trip.

☐ **D** With the RPS in a 1 out of 3 configuration, the loss of Channel 'C' results in a reactor trip.

Justification

A is correct. The K3 relay (powered from VA-30) is deenergized resulting in two TCBs opening (TCB 3 and 7). Additionally, due to the loss of another matrix relay (powered from VA-30) a contact opens in the circuitry for the' K4 relay causing TCBs 4 and 8 to open. The end result is that half of the TCBs are open, but the remaining TCBs are still providing power to the CEDMs; therefore, the reactor did NOT trip.
B is incorrect. Channel 'A' is bypassed and Channel 'C' is de-energized; however, the loss of Channel 'C' will result in 4 TCBs opening.
C is incorrect. Selected trip units on Channel 'A' are in bypass and will NOT process a trip signal for those units. The reactor will NOT trip from the trip signal processed from the loss of Channel 'C' because a signal from 2 channels is needed to cause a reactor trip.
D is incorrect. Placing Channel 'A' in bypass will NOT place the RPS in a 1 out of 3 configuration; therefore, the reactor will NOT trip when Channel 'C' is lost.

References

LP RPS-01-C, Fig. 8, RPS Drawer Power Supplies

Comments and Question Modification History

Reworded the stem such that I&C is performing a functional test on RPS Channel "A" instead of the daily RPS surveillance. The daily surveillance no longer requieres the operator to bypass any RPS trips when performing the surveillance; therefore, Distractor 'B' was implausible. 11111/08

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

Number K1.O1 **RO** 3.4 **SRO** 3.7 **CFR Link** (CFR: 41.2 to 41.9 145.7 to 45.8)

Knowledge of the physical connections and/or cause effect relationships between the RPS and the following systems: 120V vitallinstrument power system

Question #: **38** **Question ID: 8053886** ☑ **RO** ☐ **SRO** ☐ Student Handout? ☐ Lower Order?

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

Given the following conditions:
- 100% reactor power
- Inverter 2 has been deenergized in preparation for emergent repairs

The DC input breaker on Inverter 6 is inadvertently opened while hanging the clearance on Inverter 2 .

If a large break LOCA were to occur inside Containment with the plant in this configuration, which of the following would be an expected condition two minutes after the event? Assume NO operator action.

☐ **A** 'B' LPSI Pump has automatically started.

☐ **B** 'A' LPSI Pump has automatically started.

☐ **C** 'C' CAR Cooler Fan is running in fast speed.

☐ **D** 'D' CAR Cooler Fan is running in slow speed.

Justification

B is correct. Facility 1 ESAS equipment will be unaffected by the loss of Power to VA-20; therefore, 'A' LPSI Pump will automatically start.
A is incorrect. Opening the DC input breaker on Inverter 6 with Inverter 2 out will deenergize Vital AC Bus VA20, which will deenergize Facility 2 ESAS Actuation Cabinet. All Facility 2 ESAS associated equipment will be prevented from responding to conditions which would normally result in an actuation. 'B' LPSI will remain stopped until manually started by the operator
C is incorrect. The 'C'" CAR Fan is a Facility 1 Component. Facility 1 ESAS equipment will operate as designed. 'C' CAR Fan will shift to slow speed on the SIAS.
D is incorrect. Facility 2 ESAS equipment will not receive an actuation signal of any kind. 'D' CAR Fan will remain in fast speed.

References

120 VAC One-Line Diagram
ESA-01C, ESAS Lesson Text

NO Comments or Question Modification History at this time.

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

Number A3.02 RO 4.1 SRO 4.2 CFR Link (CFR: 41.7 145.5)

Ability to monitor automatic operation of the ESFAS including: Operation of actuated equipment

Question #: 39 | **Question ID:** 71905 | ☐ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

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

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

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

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

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

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

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

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

Justification

A - Correct; EOP-2532, LOCA, dictates that 2-SI-659 & 2-SI-660 are verified closed before all of the actions mentioned in the distracters. These valves being open violate CTMT integrity and offer a direct release path from CTMT to the environment. Therefore. they should be closed as soon as they are found open.
B -Wrong; Although closing the valves will isolate the HPSI minimum flow header, the accident analysis assumes these pumps will have sufficient flow to keep cool, even if being used post-SRAS.
C - Wrong; Although the RWST header isolation valves are closed, the Recirc Header isolation valves would NOT short-cycle CS through HPSI. There are valves on the discharge of the CS pumps that would do this, but they have NOT been analyzed for use and remain closed per administrative guidelines.
D - Wrong; LPSI pumps are automatically secured on a SRAS. However, if they were running due to operator action, they would challenge the other pumps based on suction flow (flow capacity of the suction strainers).

References

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

NO Comments or Question Modification History at this time.

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

Number A4.01 RO 4.5 SRO 4.8 CFR Link (CFR: 41.7 145.5 to 45.8)

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

Question #: **40** | ***Question ID:* 8000046** | ☑ ***RO*** | ☐ ***SRO*** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

Unit 2 was operating at 100% power when a rupture of an Instrument Air line causes all three RBCCW temperature control valves (TCVs) to fail full open. Service Water flow to the in-service RBCCW heat exchangers is then manually throttled to normal flow rates to minimize the effect on RBCCW heat loads.

Then, the plant trips due to a Large Break LOCA in Containment.
ALL plant equipment functions as designed, but Service Water flow to the RBCCW heat exchangers remains throttled at pre-event values.

Which one of the following statements describes how the Service Water (SW) System alignment will affect the mitigating strategy of this event?

☑ **A** Upon SRAS initiation, the RBCCW System heat exchangers will NOT provide adequate cooling for CTMT, therefore CTMT temperature and pressure will rise.

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

☐ **C** RBCCW flow through at least one SDC heat exchanger must be raised to the accident value by the time SRAS occurs or continued core cooling will be challenged.

☐ **D** To prevent exceeding system design limits, steps must be taken outside the EOPs to align both Service Water headers to one RBCCW heat exchanger.

Justification

A - CORRECT: The given conditions will result in a loss of post-accident SW flow to RBCCW, with accident level heat loads on the system will dramatically increase upon SRAS initiation.
B - WRONG: CTMT Spray will prevent CTMT pressure from exceeding the design pressure. Additionally, adequate long term cooling will not be available due to the loss of cooling to RBCCW needed for the post-SRAS environment.
C - Wrong; When SRAS occurs. RBCCW is required to provide flow to at least one CS Heat Exchanger to ensure an adequate heat sink.
D -WRONG: RBCCW design temperatures will be exceeded, based on the limited SW flow and the large increase in heat load on the GAR fans/coolers. However, the given conditions do NOT necessitate COMBINING facilities of SW as this would put the two systems in an unanalyzed alignment.

References

Millstone Unit 2 Technical Specifications Bases. Page B3/4 6-3
RCB-00-C, Pages 9 and 42
SWS-00-C, Page 33

Comments and Question Modification History

In the stem, replaced 'three' with 'inservice'; Was: Service Water flow to the three RBCCW heat exchangers... Now: Service Water flow to the in-skrvice RBCCW heat exchangers... The 'spare' heat exchanger TCV will either be full open if on minimum flow or isolated if not on minimum flow. 11/11108

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

Number K1.O1 **RO 3.5** **SRO** 3.7 **CFR Link** (CFR: 41.2 to 41.9 145.7 to 45.8)

Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: SWS/cooling system

Question #: **41** | ***Question ID:*** ***8000016*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ Student Handout? | ☐ Lower Order?

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

The plant was at 100% power when it tripped due to a Small Break Loss of Coolant Accident approximately six (6) hours ago.
Containment pressure peaked at ~12 psig, but is now reading 6.75 psig and slowly dropping.
Containment temperature is lowering appropriately with pressure.
Only Facility 1 plant components functioned as designed.
All Facility 2 components are unavailable due to Facility 2 electrical problems.
All applicable procedure steps have been successfully accomplished.

The RO has just secured Containment Spray by performing the following two actions:
- Containment Spray pump handswitch was taken to the "Close" position, then to "Trip" and then left in the "Normal-After-Trip" position.
- Containment Spray isolation valve was taken to the "Open" position, then to "Close" and then left in the "Normal-After-Close" position.

At that very moment, the break in the RCS gets worse, causing containment pressure to turn and begin slowly rising above 10 psig.

Which of the following describes actions that must be taken to ensure containment pressure control is maintained?

☐ **A** Shift the running CAR fans to "fast" speed with maximum RBCCW flow.

B Observe that ESAS automatically reinitiates containment spray flow.

☑ **C** Manually restart the containment spray pump and open the spray valve.

☐ **D** Restart containment spray flow by pushing the actuation button on C01.

Justification

C - Correct; EOP-2532, LOCA, dictates that CTMT spray should be secured once CTMT pressure drops below 7 psig. The system is secured by overriding the pump off, overriding the spray valve closed. Then, the actuating signal on ESAS is "reset" so auto actuation is again available, if needed. However, based on the steps accomplished, ESAS has NOT yet been reset. Therefore, spray flow must be manually restarted.
A - Wrong; Shifting the CAR fans to fast speed would definitely help with CTMT pressure control. However, the CAR fans are NOT designed to handle the higher density air from the LOCA in fast speed. This would overload the fans and their duct work.
B - Wrong; Because the ESAS actuation signal is still active, just over-ridden, it can NOT "reinitiate" and restart spray flow.
D - Wrong; Although the RO has secured CS flow, the actions taken so far have NOT completed the applicable procedure steps in that ESAS was NOT reset. Because the ESAS CS signal is still active, pushing the CS actuation buttons on C01 will NOT restart CS flow.

References

EOP 2532 Tech Guide, Page 109
CSS-00-C, Containment Spray. Page 14

NO Comments or Question Modification **History** at this time.

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

Number A1.02 RO 3.6 SRO 3.8 CFR Link (CFR: 41.5 / 45.5)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: Containment pressure

Question #: 42 | **Question ID: 8000017** | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

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

The plant has tripped from 100% power due to an intersystem LOCA into the RBCCW System. The RBCCW Containment Isolation Valves were successfully closed within 10 minutes after entry into Loss of Coolant Accident, EOP 2532.

The following conditions existed approximately 3 hours after the trip:
* SRAS actuated
* HPSl Pump current and flow are fluctuating.
* Containment pressure is 5 psig and slowly lowering.
* RCS pressure is 360 psia and slowly lowering.

Which of the following describes the cause of the HPSl Pump current and flow fluctuations, and the initial action that must be taken?

☐ **A** RBCCW is not providing adequate heat removal due to post-SRAS heat loading. Secure one Containment Spray Pump.

☐ **B** The HPSl Pumps are showing signs of cavitation due to low CTMT pressure. Secure one of the HPSl Pumps.

☐ **C** The loss of inventory due to the intersystem LOCA resulted in vortexing in the HPSl Pumps. Throttle HPSl injection flow.

☑ **D** The HPSl Pumps are showing signs of cavitation due to CTMT Sump clogging. Secure both Containment Spray Pumps.

Justification

D is correct. Sump clogging will cause a lower suction pressure in all the running SI pumps. A lower suction pressure will cause the HPSI Pumps to cavitate. EOP-2532 directs the CS pumps be secured (if not needed) to limit the competition for sump suction flow.
A is incorrect. Although the heat load on RBCCW dramatically increases during post-SRAS, the RBCCW system is designed for this. The intersystem LOCA would be isolated by procedure at the CTMT boundary, thereby preventing excessive heat input to RBCCW from the RCS. However, securing one CS pump would limit the heat input to the RBCCW system if this was perceived to be the cause.
B is incorrect. Although a lower Containment pressure will result in a lower suction pressure for the HPSI Pumps, the level in the Containment Sump is analyzed to be adequate to maintain the pumps in service.
C is incorrect. EOP-2532 directs the throttling of HPSI flow for sump clogging and when it is not needed. However, throttling HPSl (core cooling) flow is ONLY done if the initial action taken for this indication does NOT work.

References

EOP-2532, St. 50, Indications of CTMT Sump Clogging

NO Comments or Question Modification History **at** this time.

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

Generic KIA selected

NRC WA Generic System 2.4 Emergency Procedures /Plan

Number 2.4.47 RO 3.4 SRO 3.7 CFR Link (CFR: 41.10,43.5 145.12)

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

Question #: 43 **Question ID: 8078895** ☑ RO ☐ SRO ☐ Student Handout? ☑ Lower Order?

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

The plant is at 100% power when the extraction steam supply valve to the 2A Feed Water Heater closes.

Which one of the following describes an outcome of this event, assuming no operator action?

☑ **A** RCS Tcold on RPS will lower.

☐ **B** Delta-T power on RPS will lower.

☐ **C** Generator MWe output will lower.

☐ **D** Main Condensate Flow will lower.

Justification

A - Correct; The loss of steam supply to the feedwater heater will result in lower feedwater temperatures entering the SG. This will result in a drop in RCS Tcold.
B - Wrong; The lowering of Tcold combined with an unchanged steam demand will result in a RISE in delta-T power.
C - Wrong; The loss of steam flow to the extraction results in a gain of steam flow directly to the main turbine. This extra steam will cause generator electrical output to RISE.
D -Wrong; The extra steam going through the main turbine equates to extra water in the main condenser and a RISE in condensate flow.

References

LP FHD-01-C, Pg. 22, Potential impact on power when removing a Feedwater Heater.
0P-2318, Precaution and Notes on potential impact on power when removing isolating steam to a Feedwater Heater.

NO Comments or Question Modification History at this time.

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

Number K3.05 **RO** 3.6 **SRO** 3.7 **CFR Link** (CFR: **41.7** 145.6)

Knowledge of the effect that a loss or malfunction of the MRSS will have on the following: RCS

Question #: **44** **Question ID:** **8000050** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☐ **Lower Order?**

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

The plant is at 100% power, steady state, when an electrical fault on VA-20 causes a small fire in the VA-20 cabinet.
Within a couple seconds, the main supply breaker on VA-20 trips, deenergizing VA-20 and extinguishing the fire.

ALL other plant systems and components unaffected by the loss of VA-20 remain unchanged.

Which of the following actions is required to take control of #2 Main Feedwater Regulating Valve?

☐ **A** Place the #2 Main Feed Regulating Valve controller in "Manual" control at C-21.

☐ **B** Place the #2 Main Feed Regulating Valve to "Isolate", then "Manual" at C-10.

☑ **C** Place the #2 Main Feed Regulating Valve in "Local-Manual" control at the valve.

☐ **D** Place the #2 Main Feed Regulating Valve controller in "Manual" control at C-05.

Justification

C - CORRECT; VA-20 supplies power to the #2 Main Feedwater Regulating Valve control circuit. The loss of VA-20 will cause the #2 Main Feedwater Regulating Valve to "lock-up" as-is, which requires "local-manual" control of the valve.
A - WRONG; That is the correct method to control most components from C-21 (Hot Shutdown Panel in the West 480 VAC room), but the Main Feed Regulating Valve does NOT have any controls on C-21.
B - WRONG, That is the correct procedure for taking control of a component at C-10 (Fire Shutdown Panel in the upper 4160 KVA electrical room). However, only the #2 Aux. Feed Regulating Valve can be controlled from that panel, even though the components on C-10 are Facility 2 aligned.
D - WRONG; Although the controllers on C-05 will still be energized and appear normal (powered by VR-11 or VR-21), the control circuit in the turbine building going to the valve is deenergized with a loss of VA-20.

References

AOP-2504D, Loss of VA-20, Effect on MFRV and Required Actions

NO Comments **or Question Modification History at this time.**

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

Generic K/A Selected

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

Number 2.1.30 **RO** 3.9 **SRO** 3.4 **CFR Link** (CFR: 41.7 145.7)

Ability to locate and operate components, including local controls

Question #: **45** | ***Question ID:*** ***1000077*** ☐ ***RO*** ☐ ***SRO*** ☐ Student Handout? ☐ Lower Order?

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

The crew is in EOP 2532, Loss of Primary Coolant, responding to a Small Break LOCA complicated by a loss of Condenser vacuum.
All plant components are functioning as designed.
The US has directed a plant cooldown be initiated.
5 minutes into the cooldown annunciator window C12 on panel C05, "Condensate Storage Tank At Minimum Level" alarms.
The BOP reports that CST level has been lowering consistent with AFW usage and is presently reading 66%.

Which of the following describes an action that the US will direct based on the BOP report?

☐ **A** Transfer the contents of the Condensate Surge Tank to the Condensate Storage Tank.

☐ **B** Transfer water from the Primary Water Storage Tank to the Condensate Storage Tank.

☐ **C** Ensure maximum available makeup from the water treatment vendor and availability of firewater as the AFW suction source.

☐ **D** Halt RCS cooldown to minimize usage of AFW and immediately align the firewater header as the AFW suction source.

Justification

C - Correct; ARP C05*C12 response to CST @ minimum level alarm.
A - Wrong; Although the flow path for this transfer exists, it is administratively prohibited to avoid hotwell reject water from contaminating the CST.
B -Wrong; Although PMW is a source of water to many systems, the flow path for this transfer does NOT exist.
D - Wrong; CST volume at alarm setpoint is based on cooling down to SDC while removing decay heat for limited time, stopping cooldown lengthens decay heat removal time and may prevent reaching SDC. Also, the firewater header does NOT have to be aligned to the CST before tank level drops below ~22%.

References

ARP-2590D-047, CST At Minimum Level alarm

Comments and Question Modification History

Changed the complication in the stem from a loss of offsite power, to a loss of Condenser vacuum. Reviewer comment: too many loss of offsite power malfunctions.
Fixed typo in Distractor A. Changed 'Transferred' to 'Transfer'. 11111108

NRC K/A System/E/A System 061 Auxiliary / Emergency Feedwater (AFW) System

Number K1.05 RO 2.6" SRO 2.8* CFR Link (CFR: 41.2 to 41.9 145.7 to 45.8)

Knowledge of the physical connections and/or cause-effect relationships between the AFW and the following systems: Condensate system

Question #: **46** **Question ID:** **54554** ☑ **RO** ☐ **SRO** ☐ Student Handout? ☐ Lower Order?

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

A Loss of Normal Power has occurred, and Bus 24C is being powered from the " A Emergency Diesel Generator.
The Reserve Station Services Transformer (RSST) is now available for use.

When should the undervoltage relays on the Engineered Safeguards Actuation System be reset, to allow transferring Bus 24C to the RSST?

☐ **A** Immediately AFTER stripping all loads (except 22E) off bus 24C.

☐ **B** Immediately PRIOR to the RSST being energized from the grid.

☐ **C** Immediately AFTER paralleling the RSST with the Diesel Generator.

☐ **D** Immediately PRIOR to paralleling the RSST with the Diesel Generator.

Justification

D - Correct; With 24C being powered from the EDG due to an LNP, the ESAS Undervoltage signal would still be present. This signal prevents closing in any other source of power to the bus, other than the EDG. Therefore, the signal must be reset before the RSST breaker can be closed.
A - Wrong; This action is taken only if 24C is deenergized and is about to be repowered.
B - Wrong; This is NOT allowed as it would prevent the sequencer from slowly loading emergency equipment on the EDG if a subsequent accident resulted in a SIAS.
C -Wrong; This is NOT possible due to the UV interlock from ESAS with the EDG powering the buss.

References

EOP-2541, App 23, Attachment 23-H, Transferring 24C form EDG to RSST

NO Comments or Question Modification History at this **time.**

NRC K/A System/E/A System 062 A.C. Electrical Distribution

Number K1.02 RO 4.1 SRO 4.4 CFR Link (CFR: 41.2 to41.9)

Knowledge of the physical connections and/or cause- effect relationships between the ac distribution sys- tem and the following systems: ED/G

Question #: 47 | **Question ID:** 73066 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

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

The following conditions exist:
- Mode 2
- Reactor Startup in progress
- All switchyard breakers are CLOSED

Then, a feed control problem causes a reactor trip on low steam generator level.

Based on EOP 2525, Standard Post Trip Actions, which one of the following describes the required actions to ensure the switchyard and transformer yard are properly configured?

☐ **A** Verify the 15G-2X1-4, Motor Operated Disconnect, is open and the 15G-8T-2 and 15G-9T-2, Generator Output Breakers, are closed.
Verify all facility 1 and 2 electrical buses energized

☐ **B** Verify the 15G-8T-2 and 15G-9T-2, Generator Output Breakers, are open and the 15G-2X1-4, Motor Operated Disconnect, is closed.
Verify all facility 1 and 2 electrical buses energized

☐ **C** Verify the 15G-2X1-4, Motor Operated Disconnect, and the 15G-8T-2 and 15G-9T-2, Generator Output Breakers, are closed
Verify all facility 1 and 2 electrical buses energized

☐ **D** Verify the 15G-8T-2 and 15G-9T-2, Generator Output Breakers, opened automatically
Verify diesel generators powering both 24C and 24D

Justification

A - Correct; With the ring bus closed, the 8T & 9T are closed and should remain that way. The 15G-2x1-4 must, therefore, be open because the Main Generator is off line.
B -Wrong; The 8T & 9T should remain closed because NO trip signal will be generated in this configuration. In MODE2, the MOD, 15G-2X1-4, is open.
C - Wrong; Unit 2 does NOT have a Main Generator output breaker, therefore, the 15G-2X1-4 must be open if the ring bus is closed.
D - Wrong; The 8T & 9T do NOT get an automatic trip signal because the main generator does NOT get a trip signal from the main turbine.

References

Training Diagram for 345 KV

Comments and Question Modification History

Added "Based on EOP 2525, Standard Post Trip Actions" to provide a basis for the required actions.
Discussed adding a failure of the Main Steam Stop valves to close which, under normal power conditions, would prevent the 15G-8T-2 and 15G-9T-2, Generator Output Breakers from opening. In this case the 15G-8T-2 and 15G-9T-2, Generator Output Breakers should NOT open. Therefore it was deemed unnecessary to add the malfunction.
Added noun names to the breakers and disconnect. 11111108

NRC K/A System/E/A System 062 A.C. Electrical Distribution

Number A4.01 RO 3.3 SRO 3.1 CFR Link (CFR: 41.7 145.5 / to 45.8)

Ability to manually operate and/or monitor in the control room: All breakers (including available switchyard)

Question #: **48** **Question ID: 8073622** ☑ RO ☐ SRO ☐ Student Handout? ☐ Lower Order?

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

The plant has just tripped from 100% power and the following conditions now exist:
- All breaker indicating lights for Bus 24C are deenergized.
- Breaker indicating lights for TCBs #1 and #5 are deenergized.
- The appropriate breaker indicating lights for all other buses are lit.
- Buses 24A and 24C are deenergized.

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** #1 Atmospheric Dump Valve controller on C05 is deenergized.

☐ **D** The " A and "C" RCPs are running without cooling water.

,Justification

D - Correct; This is the indication for the loss of DV-10. The loss of DC (control power) will also cause a loss of 24A & 24C on the trip, because the RSST-24C breaker and the "A" DIG output breaker cannot close. With no facility 1 power there is no facility 1 RBCCW, so the two RCPs are running without cooling water and should be immediately tripped manually.
A - Wrong; The Condensate Pumps and RCPs are powered from Buses 25A & B, which still have power. This would be true if the DC bus 201A was lost, but that would cause the breaker lights on 24A to also deenergize.
B -Wrong; The "A" EDG will start on a loss of DC (DV-10) with only "overspeed" protection. The "emergency" trips are NOT available. This would be the expected condition on a normal loss of offsite power start.
C -Wrong; The #1 ADV controller is powered by VA-10, NOT DV-10. Vital control power bus VA-10 is still being powered from 201A.

References

AOP-2506A, Loss of DV-10 Load List

~ NO Comments or Question Modification History at this time.

NRC K/A System/E/A System 063 DC Electrical Distribution System

Number A4.01 RO 2.8* SRO 3.1 CFR Link (CFR: 41.7145.5 to 45.8)

Ability to manually operate and/or monitor in the control room: Major breakers and control power fuses

Question #: **49** | **Question ID:** **56971** | ☐ **RO** | ☐ **SRO** | ☐ Student Handout? | ☑ Lower Order'?

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

Following a LNP automatic start of the 'B' DG, a "DIESEL GENERATOR 13U TROUBLE alarm is received. A PEO dispatched to investigate the alarm reports that the "JACKET COOLING TEMP HIGH" alarm is active and 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 ruptured in the overhead by EBFAS Fans.

☐ **D** B DG SW supply is inadvertently cross tied with the A DG SW supply.

Justification

B - Correct; The SW supply valve will automatically open when the DG is started, but the bypass valve does NOT auto open or close.
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.
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.

References

ARP-2591B-009, "B" EDG Jacket Coolant Temp. High alarm response.

$Commentsand Question Modification History

Reworded Distractor 'C' to ensure the break is in the supply line upstream of the flow instrument. This still makes the distractor plausible if the examinee is not familiar with the location of the flow instrument. 11/11/08

NRC K/A System/E/A System 064 Emergency Diesel Generators (ED/G)

Number K1.02 RO 3.1 SRO 3.6* **CFR** Link (CFR: 41.2 to 41.9 145.7 to 45.8)

Knowledge of the physical connections andior cause- effect relationships between the ED/G system and the following systems: D/G cooling water system

Question #: **50** | *Question ID:* **8000019** | ☑ **RO** | ☐ **SRO** | ☐ Student Handout? | ☐ Lower Order?

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

An Aerated Liquid Radioactive Waste (ALRW) discharge was started 15 minutes ago with the following conditions:

- All ALRW components considered operating as designed
- Radiation Monitor Alarm Setpoint = 4.5 X 10 +5 cps
- Radiation Monitor Fail Setpoint = 1 X 10 +2 cps
- Radiation Monitor reading in the first five (5) minutes was ~ 3.4 X 10 +4 cps

An operator checking the progress of the discharge at this time observes the following:

- Radiation Monitor output reading is fluctuating between ~ 3 X10 +4 cps and ~ 5 X 10 +2 cps.
- Discharge flow rate is stable.
- NO alarms are active on any applicable panel.

Based on the observed conditions, which of the following describes the discharge status and required actions?

☐ **A** The discharge may continue without stopping if Chemistry draws a second sample and the new results are comparable to the initial sample results.

☐ **B** The discharge may continue without stopping if the ALRW discharge radiation monitor response is the same when a source check is performed.

☑ **C** The ALRW discharge radiation monitor is NOT operable; the discharge cannot continue with the existing permit and must be immediately secured.

☐ **D** ALRW discharge flow rate is too high for the radiation monitor; the discharge flow must be reduced to continue discharging with the existing permit.

Justification

C - Correct; The purpose of the 15 minute check is to verify the rad. monitor is operable by doing a comparison with previous indicated rad. levels. A fluctuating readout is NOT expected as the levels should compare with those seen at the start of the discharge.
A - Wrong; The discharge can be made with an inoperable rad monitor, provided a second sample is taken and analyzed. However, this must be done BEFORE the discharge is started as the permit must state the discharge is being accomplished in this manner.
B - Wrong; A source check is performed to verify the rad monitor is functioriing to sense a radioactive substance. However, this test is done before the discharge is started, to verify initial rad monitor operability, and can NOT be done while a discharge is in progress.
D - Wrong; The rad monitor gets a steady slip stream sample flow via a separate sample pump. Discharge flow rate would NOT affect the monitors ability to accurately measure radiation levels.

References

SP-2617A, Radioactive Liquid Waste Discharge Discussion and Channel Check

(NOComments or Question Modification History at this time.

NRC K/A System/E/A System 073 Process Radiation Monitoring (PRM) System

Number **A2.01** RO 2.5 SRO 2.9* CFR Link (CFR: 41.5 143.5 145.3 145.13)

Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to cor- rect, control, or mitigate the consequences of those malfunctions or operations: Erratic or failed power supply

Question #: **51** | **Question ID:** **1000069** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The plant was tripped and EOP 2536, ESDE, entered after EOP 2525 due to a large Main Steam Line Break on #2 SG inside containment.
The following conditions existed at the time:
- #2 SG has just blown dry but has not yet been isolated.
- RCS temperature and pressure have been stabilized with Th subcooled margin at 94° F.
- Containment pressure is 21 psig and lowering.
- Containment temperature is 260 °F and lowering.
- There are NO indications of any fuel clad failures.

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

Which of the following radiation monitor indications would change if the only additional casualty was a tube rupture on #2 SG?

☐ **A** Containment Atmospheric Radiation Monitors

☐ **B** Steam Jet Air Ejector Radiation Monitor

☐ **C** Facility 2 Main Steam Line Radiation Monitor

☐ **D** Refueling Floor Area Radiation Monitor

Justification

D - CORRECT: With low RCS activity and the ruptured SG already faulted this RM and the personnel access hatch area RM are the only RMs capable of alarming. The STA will note this problem when performing the Safety Function Status Check for EOP-2536.
A - WRONG: Containment Atmospheric Rad. Monitors sampling path was isolated on the CIAS triggered by high containment pressure.
B -WRONG: Ordinarily one of the first indications of a SGTR. However, the MSIVs closed on the MSI from high containment pressure, therefore, the SJAE do NOT have a steam supply.
C - WRONG: The location of RM and 30 mr/hr alarm setpoint would require significant clad failure for alarm to come in (design function).

References

EOP-2536, Safety Function Status Check
Main Steam System Diagram

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 073 Process Radiation Monitoring (PRM) System

Generic KIA **Selected**

NRC KIA Generic **System** 2.1 Conduct of Operations

Number 2.1.31 **RO** 4.2 **SRO** 3.9 **CFR Link** (CFR: 45.12)
"Ability to locate control room switches, controls and indications and to determine that they are correctly reflecting the desired plant lineup."

Question #: **52** **Question ID: 71544** ☑ **RO** ☐ **SRO** ☐ Student Handout? ☐ Lower Order?

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

The plant is operating in MODE 1 at 100% power with the following conditions:
- Injection temperature is 73°F
- " A and "C" Service Water Pumps are supplying Facility 1 and 2, respectively.
- Bus 24E is aligned to bus 24C.

Then, the 'A' Service Water Pump trips on overload. The BOP attempted to start the "B" Service Water Pump on Facility 1, but the breaker would NOT close. Subsequently, the " A RBCCW Header high temperature alarm annunciates. Within a few minutes, the BOP informs the US that the " A RBCCW heat exchanger outlet temperature is reading 121°F and rising.

Which of the following describes the minimum procedurally required actions for these conditions?

☐ **A** Log into Tech Spec 3.0.3 and restore the Facility 1 Service Water Header within one (1) hour or commence a plant shutdown.

☐ **B** Open both Service Water cross tie valves to allow the "C" Service Water Pump to supply both headers and continue operation.

☐ **C** Log into the applicable Tech Spec for loss of Facility 1 RBCCW and Facility 1 Service Water Headers and commence a Rapid Downpower.

☑ **D** Place the Facility 1 RBCCW Pump in PULL-TO-LOCK, refer to AOP 2564, Loss of RBCCW, manually trip the plant and go to EOP-2525.

Justification

D - CORRECT; The design temperature of the RBCCW system is 120°F. In accordance with AOP 2565 (Loss Of Service Water), Section 10, if RBCCW heat exchanger outlet temperature approaches 120°F (or higher) and restoration is NOT imminent, the associated RBCCW pump must be tripped. Also, AOP-2564, (Loss of RBCCW) gives guidance on RBCCW Heat Exchanger outlet temperature of >120 °F, which requires a plant trip.
A - WRONG; AOP-2565 and AOP-2564 give guidance for logging into various TSAS due to the loss of a Service Water and RBCCW headers. This could infer Tech. Spec. 3.0.3 applies due to the applicability of multiple Tech. Spec. actions.
B - WRONG; Procedural guidance exists for this action, but it is administratively prohibited in this mode of operation.
C - WRONG; When RBCCW header temperature exceeds 120 °F, the header is inoperable. Ordinarily, an inoperable cooling system header would require logging into the applicable TSAS. However, at this temperature the RBCCW header is NOT considered just inoperable, but LOST, and the appropriate actions must be taken.

References

AOP-2565, Loss of Service Water, St. 10.3
AOP-2564, Loss of RBCCW, St. 3.3.1 (Contingency)

Comments and Question Modification History

Changed stem to include that the BOP attempted to start the "B" Service Water Pump on Facilityl, but the breaker would NOT close. Previous wording was to ambiguous to ensure the examinee understood that the "B" 'Service Water Pump was NOT available.
11/11/08

NRC K/A System/E/A System 076 Service Water System (SWS)

Number K1.09 RO 3.0* SRO 3.1' CFR Link (CFR: 41.2 to 41.9 145.7 to 45.8)

Knowledge of the physical connections and/or cause-effect relationships between the SWS and the following systems: Reactor building closed cooling water

Question #: 53 | **Question ID:** 8053533 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 4 | ☑ Selected for Eixam | **Origin:** Mod | ☐ Past NRC Exam?

The plant was operating at 100%, all components are operating normally with the following Instrument Air alignment:
- The "D" Instrument Air (IA) compressor is aligned to operate in lead.
- The "E" and "F" Air Compressors are in Standby.

Then, the reactor is manually tripped due to a state wide blackout and loss of the grid. On the trip, bus 24C deenergizes due to a bus fault.
All other plant equipment responded normally to the existing conditions.

With NO OPERATOR ACTION, what is the status of the Instrument Air System?

☐ **A** Only Unit 3 is available to supply Station Air, which must then be cross-tied to the IA header.

☐ **B** The "F" IA compressor is running or will automatically start on low IA header pressure.

☐ **C** The "F" IA compressor is available, but will NOT run until manually started locally.

☐ **D** Only the "D" IA compressor is running, with "F" IA Compressor available for backup.

Justification

C - CORRECT; "F" IA compressor is the only compressor that has power. However, the "F" IA Compressor (IAC) will NOT auto-start even after the bus is reenergized by the EDG. Both "E" & "F" IACs must be given a local start signal to reset them back in "auto" mode after a loss of power.
A - WRONG; This was correct before the new "vital" IACs ("E" & "F") were recently installed. The "F" IAC has power available, but requires local operator action to supply the IA header.
B - WRONG; The "F" IAC requires local operator action to restore it to "auto start" mode.
D - WRONG; "D" IA Compressor is powered from 22C (Non-Vital480 VAC), which is deenergized because of the 24C bus fault.

References

EOP-2525, St. 18, Subsequent Action for IA restoration.
OP-2328B, Discussion on IAC Auto Start Requirements

Comments and Question Modification History

Changed the stem to state that both "E" and "F" Air Compressors are in standby. Previous wording had one compressor available and one in standby. New wording more accurately reflects actual conditions.
Changed the wording in Answer C from '...reset locally' to '...started locally'. More acturately reflects the action taken. 11111/08

NRC K/A System/E/A **System** 078 Instrument Air System (IAS)

Generic WA Selected

NRC KIA Generic **System** 2.1 Conduct of Operations

Number 2.1.27 **RO** 2.8 **SRO** 2.9 **CFR Link** (CFR: 41.7)
Knowledge of system purpose and or function.

Question #: **54** | **Question ID:** **54821** | ☑ **RO** | ☐ **SRO** | ☐ Student Handout? | ☐ Lower Order?

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

Core alterations are in progress on Unit 2. The " A Steam Generator upper manways have been removed to perform an inspection of the moisture separators. #2 S/G is intact.
It was subsequently discovered that a Main Steam Safety Valve had been removed from the " A Main Steam Header for maintenance, instead of removing a safety from the intended "B" Main Steam Header.

What action(s), if any, is required in response to this discovery?

☐ **A** Immediately suspend all movement of irradiated fuel assemblies in CTMT.

☐ **B** Immediately verify plant ventilation is maintaining a negative pressure in CTMT.

☐ **C** No action is required provided RCS temperature is maintained < 200 °F.

☐ **D** No action is required if a blue cloth FME cover is in place of the removed valve.

Justification

A - CORRECT; T.S. 3.9.4. - With manways removed, the removed safety allows a direct path for radioactivity release to the atmosphere if fuel damage were to occur.
B - WRONG; OK if "Core alterations" were NOT in progress.
C -WRONG; This assumes the requirement for CTMT Integrity to perform a plant heat up.
D - WRONG; Although FME is a concern, this assumes concern is for foreign object damage only. An FME cover is NOT adequate for CTMT atmosphere isolation.

References

Main Steam Diagram showing path through CTMT wall.
Tech Spec Action Required for violation of CTMT Integrity.
Tech Spec Definition of "Core Alteration" and CTMT Integrity TS.

Comments and Question Modification History

Added the sentence, "#2 SIG is intact." To provide information as to status of #2 SIG. For clarity.
Deleted reference to the the Auxiliary Building PEO and added that it was a subsequent discovery that a safety had been removed from "A" Main Steam header instead of the " B Main Steam Header. The discovery would not likely be from the Aux Building PEO.

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

Number K3.01 RO 3.3* SRO 3.7' CFR Link (CFR: 41.7 145.6)

Knowledge of the effect that a loss or malfunction of the containment system will have on the following: Loss of containment integrity under shutdown conditions

Question #: **55** | **Question ID: 8000020** | ☑ **RO** ☐ **SRO** | ☐ Student Handout? | ☐ Lower Order?

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

The plant is at 100% power, steady state, with the following additional conditions:
- RCS leakage has risen over the last couple days to approximately 1.5 gpm.
- HP and Operations are performing an Emergency Containment Entry to investigate the source of leakage.
- All other plant conditions and systems are normal.

The team has entered containment (CTMT) and observes that 2-CH-442, the Letdown Header isolation to the RCS, has a small body-to-bonnet leak.
While exiting CTMT, the door interlock mechanism fails, causing both doors to jam open about 50%.
Various ventilation radiation monitors are now beginning to slowly rise.

Which one of the following describes a consequence of the broken CTMT air lock and an applicable mitigating strategy?

☐ **A** Radiation leaking from CTMT to the Enclosure Building is a "ground release". Main Exhaust must be aligned to Millstone Stack.

☑ **B** Radiation is leaking out of CTMT to the Enclosure Building. EBFAS must be initiated to minimize the release to the environment.

☐ **C** CTMT leakage is now much greater. All Enclosure Building doors must remain closed to ensure CTMT INTEGRITY is maintained.

☐ **D** The CTMT Barrier is lost and the RCS Barrier is degraded. A plant trip is required, followed by an immediate cooldown to Mode 5.

Justification

B - Correct: With the rise in leakage from CTMT, an Enclosure Building (EB) Filtration Actuation Signal will start the EB Fans, filter the radiation released from CTMT with HEPA filters and realign EB ventilation to the millstone stack to eliminate the "ground release" effect.
A - Wrong; main exhaust can NOT be aligned to the millstone stack. It discharges only to the MP2 stack, which is considered a "ground release"
C - Wrong; CMTM Integrity is violated as long as the air lock doors are not closed and sealed. Integrity can NOT be re-established regardless of actions taken to secure the EB.
D - Wrong; The barriers are lost, but a plant trip is NOT required and not necessarily conservative. A trip from 100% power will put a lot of stress on plant components, including the leaking valve, and should not be done when a Rapid Downpower is an option.

References

OP-2314G, Enclosure Building Filtration System Discussion
EBFAS Diagram

Comments and Question Modification History

Much discussion on whether or NOT this is RO knowledge. Based on KIA importance - yes.

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

Number A2.05 RO 2.9 SRO 3.9 CFR Link (CFR: 41.5 / 43.5 145.3 145.13)

Ability to (a) predict the impacts of the following malfunctions or operations on the containment system-and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations Emergency containment entry

Question #: 56 | **Question ID: 8080579** ☑ **RO** ☐ **SRO** ☐ Student Handout? ☐ Lower Order?

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

The reactor is at Middle of Life (MOL), and has tripped from 100% power equilibrium conditions. .The cause of the trip has been determined and corrected with NO other maintenance issues.
Critical rod position has been calculated for a reactor startup eight (8) hours after the trip and the RCS boron concentration has been adjusted per the ECP.

Which of the following conditions would cause the actual critical rod position to be lower than the predicted critical rod position?

☐ **A** Startup is delayed for an additional six (6) hours beyond the time used for the original ECP calculations.

☐ **B** When Borating to the RCS, CVCS was accidentally aligned to the " A BAST, instead of the RWST.

☐ **C** While performing the reactor startup, the #1 Steam Generator steam flow transmitter begins to fail high.

☐ **D** Beginning of Life curves where used by mistake in performing the ECP calculations used for the startup.

>Justification

A - Correct; Delaying the startup an additional 4 hours will put the startup well after Xenon has peaked and the concentratlon has lowered below what levels would have existed 10 hours after the trip. The lower Xenon concentratlon amounts to a positive reactivity addition, resulting in a lower CEA height for criticality.
B - Wrong; The BAST have a higher concentratlon of boron than the RWST, which results in an Increase in boron in the RCS and a negative reactivity addition, or a higher CEA height for criticality.
C -Wrong; At greater than 15% power, this would cause the Main FRVs to open and over-feed the Steam Generator, causing a drop in RCS temperature and a positive reactivity addition. But at the power level that a reactor startup is performed, steam flow has NO affect on the amount of feed water going to the SG
D -Wrong; BOL curves would have assumed a much greater excess reactivity present in the core This would result in the ECP requiring a higher RCS boron concentration.

'References

Reactor Engineering, Curve and Data Book, MOC Life Post-Trip Xenon Decay

:Comments and Question Modification History

If peak Xenon is assumed to occur 10 hours post trip, then a delay of 4 hours (Answer 'A') will result in the same Xenon concentration as the original ECP. If the startup is delayed 6 hours, Xenon is less than the ECP.
Changed distractor "B" to state that boration was from the BAST instead of the RWST. Originally was reversed, making the distractor a correct answer. 11/11/08

NRC K/A System/E/A System 001 Control Rod Drive System

Number K5.13 RO 3.7 SRO 4.0 CFR Link (CFR: 41.5/45.7)

Knowledge of the following operational implications as they apply to the CRDS: Effects of past power history on xenon concentration and samarium concentration

Question #: **57** | **Question ID:** 81726 | ☑ **RO** | ☐ **SRO** | ☐ Student Handout? | ☐ Lower Order?

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

The plant is at 98% power with Group 7 CEAs at 170 steps withdrawn.
The Plant Process Computer (PPC) has just been lost (totally shutdown).

Which one of the following describes when all rod motion will automatically stop, if Group 7 rods are now withdrawn in Manual Group mode?

☐ **A** Manual Group rod motion will stop when the FIRST rod in the group reaches the Upper Electrical Limit (UEL).

☑ **B** Manual Group rod motion will stop when ALL the rods in the group reach their Upper Electrical Limit (UEL).

☐ **C** Manual Group rod motion will stop when the LAST rod in the group reaches the Upper Core Stop (UCS).

☐ **D** Manual Group rod motion will stop when the FIRST rod in the group reaches the Upper Core Stop (UCS).

Justification

B - CORRECT; UCS is driven by the PPC, therefore, the CEAs will only stop when they reach the UEL.
A - WRONG; The UEL does NOT input into the CEA Group controllers-in the CEDS Logic Cabinets. It inputs directly into the Individual CEA controllers for the purpose of stopping ALL withdraw commands before the CEA reaches the mechanical limit of the CEDM.
C - WRONG; This is when group withdrawal would normally be stopped by the Upper Core Stop. However, the UCS is driven by the PPC and is, therefore, unavailable.
D - WRONG; This is a common misconception of when group motion would be stopped by the Upper Core Stop, based on an UCS setpoint of 177 PULSES (RPI from the PPC) and the a UEL setpoint of 180 steps (RPI from reeds). When rods are being withdrawn and a slight misalignment within the group exists, often the difference between the two readings makes it appear that rod motion does not stop until at least one CEA has reached its UEL.

References

OP-2302A, Pg. 48, Attachment 5

Comments and Question Modification History

Reworded question to improve legibility.

NRC K/A System/E/A System 014 Rod Position Indication System (RPIS)

Number K4.01 **RO** 2.5' **SRO** 2.7* CFR Link (CFR: 41.5 / 45.7)

Knowledge of RPIS design feature(s) and/or interlock(s) which provide for the following: Upper electrical limit

Question #: 58 | **Question ID: 8064354** | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

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

A plant down power is in progress with present power level at ~12% and dropping slowly.
Three of the four RPS Linear Power Range bistables have been reset (LEDs have gone out). However, the Channel "D" power range bistable will NOT reset (the LED remains lit and is not blinking).
I&C investigation reveals the RPS Channel "D" level 1 bistable is failed in the "armed" state, but all other components of Channel "D" are operating normally and are expected to continue functioning as designed.

Which one of the following describes the effect of the RPS Level 1 Bistable's existing status?

☐ **A** The Channel "D" trip signal from the "D" Turbine Stop valve is blocked.

☐ **B** One of the four CEDS bus Undervoltage Relays is failed or tripped.

☐ **C** Tripping the Main Turbine for the plant shutdown will trip the reactor

☐ **D** The Local Power Density trip on RPS Channel "D" is still armed.

Justification

D - Correct; Level 1 Bistables will "reset" below 15% NI Dower as sensed by the Linear channels to bypass the turbine trip and LDP trip for that channel of RPS. Therefore, they are still armed for this channel.
A - Wrong; The RPS channels sense closure of the main turbine "control" valves, NOT the stop valves. It is a failure of the "stop" valve close signal lhat would prevent the 8T & 9T from getting a trip signal.
B -Wrong; CEDS undervoltage relays have no direct input to RPS. When RPS trips the reactor, the CEDS UV deenergize and send a signal to the turbine control system to trip the turbine.
C - Wrong; RPS channel "D" will process the turbine trip and trigger. However, because of the 214 logic, the reactor will NOT trip.

References

OP-2205, R-14, Pg 14 & 15

Comments and Question Modification History

Changed Distractor "A" from "...signal to the 15G-8T-2 and 9T-2 is bloscke ' to '...signal from the " D Turbine Stop valve is blocked.' This is a more plausible distractor. 11/11108

NRC K/A System/E/A **System** 015 Nuclear Instrumentation System

Number K6.04 **RO** 3.1 **SRO** 3.2 **CFR Link** (CFR: 41.7 145.7)

Knowledge of the effect of a loss or malfunction on the following will have on the NIS: Bistables and logic circuits

Question #: **59** **Question ID: 8600105** ☑ **RO** ☐ **SRO** ☐ Student Handout? ☐ Lower Order?

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

The plant is in normal operation at 100% power with all systems and components aligned normally and functioning as designed.

Then, the Loop 1Thot input to the Reactor Regulating System suddenly fails low.

Which of the following actions are required to ensure Pressurizer level is maintained on program for the actual plant conditions?

☐ **A** Verify the Plant Process Computer has "bypassed" the failed Thot input and calculated outputs are at pre-failure levels.

☑ **B** Verify the Foxboro IA program has "bypassed" the failed Thot input and calculated outputs are at pre-failure levels.

☐ **C** To ensure proper operation in the event of a trip, select "Local-Setpoint" on the selected pressurizer level controller.

☐ **D** To ensure proper operation in the event of a trip, transfer PZR level control to Channel 'X' on the Foxboro IA controller screen.

Justification

B - Correct; the Foxboro IA will automatically de-select an input that is failed out-of-range and use only the other loops Thot for the calculation of pressurizer level program setpoint and steam dump valve auto demand setpoint. However, it should be verified that this occurs per the design.
A - Wrong; Although the Foxboro IA is controlled from a terminal and screen used for interface with the PPC, the program is running on a totally different computer system. Also, the Foxboro IA sends data to the PPC for use and display, NOT the other way around.
C - Wrong; This will PREVENT the RRS/Foxboro IA from controlling pressurizer level as designed in the event of a plant trip.
D - Wrong; This action may be warranted if the Foxboro IA Tavg signal were to fail low with the failed input. However, when a loop temperature fails low, the Foxboro IA automatically bypasses the bad input; therefore, Tavg and PZR level control will NOT be affected.

References

RRS-01-C, Reactor Regulating System, Rev. 3, Ch. 3, Page 18

Comments and Question **Modification** History

Changed the failure on Reactor Reg from 1200°F (high) to a low failure. If the temperature fails high and the Foxboro IA does not bypass the failed input, then nothing happens; if the the temperature fails low and the Foxboro IA does not bypass the failed input. then operator intervention is required. More challenging.
Changed Distractor 'D' to transfer PZR level control to channel 'X' instead of steam dump controls. Makes the distractor more plausible.

NRC K/A System/E/A System 016 Non-Nuclear Instrumentation System (NNIS)

Number A4.01 RO 2.9* SRO 2.8* CFR Link (CFR: 41.7 145.5 to 45.8)

Ability to manually operate and/or monitor in the control room: NNI channel select controls

Question #: 60 | Question ID: 8100018 | ☐ RO | ☐ SRO | ☑ Student Handout? | ☐ Lower Order?

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

The following conditions exist:

- The plant has tripped due to a Loss of Coolant Accident.
- VA-20 was lost on the trip and cannot be restored.
- The RVLMS indicates 0% vessel level [Note: The 7% Level Indication (point #8) has been jumpered out on both channels due to instrument failure.]
- Containment pressure is 35 psig and stable.
- The crew has just completed the Diagnostic Flow Chart.

Which of the following sets of data would provide definite indication that the core is actually uncovered, likely resulting in core damage, and what action must be taken to mitigate the effects of this condition?

☐ **A** Pressurizer pressure: 250 psia; Maximum HJTC temperature: 450°F; CET Max: 405°F; CET High: 400°F
Reduce RCS pressure to raise LPSI flow in accordance with EOP 2532, Loss of Coolant.

☐ **B** Pressurizer pressure: 50 psia; Maximum HJTC temperature: 350° F; CET Max: 285°F; CET High: 280°F
Start the Facility 2 Safety Injection Pumps in accordance with EOP 2532, Loss of Coolant.

☑ **C** Pressurizer pressure: 250 psia; Maximum HJTC temperature: 450°F; CET Max: 425°F; CET High: 420°F
Start the Facility 2 Safety Injection Pumps in accordance with EOP 2532, Loss of Coolant.

☐ **D** Pressurizer pressure: 50 psia; Maximum HJTC temperature: 350° F; CET Max: 415°F; CET High: 410°F
Reduce RCS pressure to raise LPSI flow in accordance with EOP 2532, Loss of Coolant.

Justification

C - CORRECT; With 250 psia, and CET High at 420°F, conditions indicate superheat at the top of the core. Superheated conditions at the top of the core are indicative of core uncovery. The only way to cover the core is to increase Safety Injection flow. This is accomplished by manually starting the Facility 2 Safety Injection Pumps which failed to automatically start due to the loss of VA-20.
A - WRONG; RVLMS @ 0% only means the lowest RVLMS thermocouple (at 7%) is uncovered, the heated junction thermocouple (HJTC) would be expected to read >200° F above saturation temperature of 511° F;
B - WRONG; RVLMS @ 0% only means the lowest RVLMS thermocouple (at 7%) is uncovered, PIT relationship is saturated;
D - WRONG; HJTC reading is within reason for uncovered in a saturated environment, P/T relationship is saturated.

References | **Provided**

EOP-2541 Pressure/Temperature Requirements.
[Provide Steam Tables during exam]

Comments and Question Modification History

Added 'on both channels' to stem to clarify that both channels have the lower probe jumper out. 11111108

NRC K/A System/E/A System 017 In-Core Temperature Monitor System (ITM)

Number A2.02 RO 3.6 SRO 4.1 CFR Link (CFR: 41 5 143.5 145.3 145.5)

Ability to (a) predict the impacts of the following malfunctions or operations on the ITM system; and (b) based on those predictions, use procedures to cor- rect, control or mitigate the consequences of those malfunctions or operations: Core damage

Question #: **61** | ***Question ID:*** ***8000021*** ☑ ***RO*** ☐ ***SRO*** ☐ Student Handout? ☑ Lower Order?

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

The plant tripped due to a LOCA several hours ago and the following conditions now exist:

- EOP-2532 in progress
- Containment pressure = 2.5 psig and lowering slowly
- SIAS, CIAS, EBFAS, MSI fully actuated and verified
- CSAS secured based on improving containment conditions
- Containment Hydrogen purge in progress
- All other plant systems and components functioning as designed

Then, degrading conditions in the RCS causes Containment pressure to start rising.

Which of the following describes how the Containment Hydrogen Purge duct work will be protected from an over-pressure condition?

☐ **A** The Hydrogen Purge Dampers must be manually closed from C-01.

☐ **B** The Hydrogen Purge Dampers must be closed locally by a PEO.

☐ **C** Containment purge will be automatically isolated by re-actuation of a CSAS.

☐ **D** Containment purge will be automatically isolated by re-actuation of a CIAS.

Justification

A - Correct; The isolation valves must be overridden open and, therefore, must be closed by re-operation of their control switches.
B - Wrong; The valves do NOT have to be locally operated based on the given conditions. Although, local-manual operation of these valves is often practiced as an operator task (location allows for easy access) it is NOT the preferred method due to ALARA.
C - Wrong; Although CSAS is "reset" when spray is secured, and it will automatically reactuate on rising CTMT pressure, this signal will NOT automatically close the purge isolation valves.
D - Wrong; If CIAS where to reactuate on rising CTMT pressure it would automatically close the purge isolation valves. However, the given conditions do NOT state that the CIAS signal has been reset and, unlike the direction given to secure and reset CSAS, EOP-2532 does NOT direct CIAS be reset on improving CTMT conditions.

References

LP CSS-01-C, CTMT Purge Isolation Valves w/ CIAS.

Comments and Question Modification History

Answer 'A' and Distractor 'B' Changed Containment Purge Isolation Valves to Hydrogen Purge Damper. Correct terminology. 11/11/08

NRC K/A System/E/A System 028 Hydrogen Recombiner and Purge Control System (HRPS)

Number K1.01 RO 2.5* SRO 2.5 CFR Link (CFR: 41.2 to 41.9 145.7 to 45.8)

Knowledge of the physical connections and/or cause-effect relationships between the HRPS and the following systems:
Containment annulus ventilation system (including pressure limits)

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

Question #: **62** | ***Question ID:*** **8000022** | ☑ ***RO*** | ☐ ***SRO*** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

During refueling operations, the RO notices that the audible count rate from the Nuclear Instruments (NIs) appears to have suddenly started to rise.
The US directs the RO to validate the audible count rate change, before initiating any mitigating actions.

Which of the following would immediately validate the audible count rate change heard in the control room?

☐ **A** Monitor NI Safety Channel indication on RPS or C04.

☐ **B** Request audible count rate indication from the Refuel Machine.

☑ **C** Compare wide range channel trends on the recorder on C04

☐ **D** Monitor NI Control Channel indication on RPS or C04.

Justification

C - Correct; Any of the wide range channels can be selected to plot on the C04 recorder that is dedicated for those instruments This would give a quick indication of whether the count rates are actually rising.
A - Wrong; The Safety Channel NIs are in the "power" range and were NOT designed for Indication of count rate changes at this level.
B -Wrong; The audible count rate indication heard on the refuel machine is from the same wide range channel heard in the control room. Checking this indication would only verify the audible control circuits both work, NOT that the instrument feeding them is reading correctly.
D - Wrong; The Control Channel NIs are also in the "power" range and were NOT designed for indication of count rate changes at this level.

References

LP Wide Range NI Diagram
Excerpt from OP-2202, Rx S/U directing use of various Wide Range NI indications

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 034 Fuel Handling Equipment System (FHES)

Number A4.02 **RO** 3.5 **SRO** 3.9 **CFR** Link (CFR: 41.7 145.5 to 45.8)

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

Question #: **63** **Question ID: 8000023** ☑ RO ☐ SRO ☐ Student Handout? ☑ Lower Order?

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

Which of the following plant scenarios would cause the greatest change in the numerical value of the Moderator Temperature Coefficient (MTC)?

☐ **A** Raising plant power from 1x10E-4% to 1% during the initial reactor startup directly after a refueling outage.

☐ **B** Raising output from 600 MWe to 900 MWe, recovering from an Emergency Generation Reduction using steam dumps.

☐ **C** Returning RCS temperatures, from Coastdown values to normal program values, for end-of-cycle shutdown.

☑ **D** Returning plant power to 100% from that required by Tech. Specs. due to a dropped CEA, at Middle-Of-Life.

Justification

D - Correct; reactor power must be lowered to less than 70%, and turbine load with it, to comply with maximum allowed power level with a dropped CEA. Boron injection must be used (due to the dropped CEA) and at MOL conditions, this will raise RCS boron concentration enough to effect the value of MTC. Once the CEA is recovered, the added boron must be diluted out to return power to 100%.

A - Wrong; The amount of dilution required to raise power to 30% at BOL is NOT significant enough to change MTC very much.

B - Wrong; Reactor power and RCS temperature is NOT changed, by procedure, in an Emergency Generation Reduction event.

C - Wrong; RCS boron concentration is NOT altered during this event. RCS temperature is raised lo program by lowering turbine load.

References

AOP-2556, Power Reduction Requirement

Comments and Question Modification Historv

Changed Distractor 'A' changed 0-30% power to 1x10E-4% power to 1%. To show a wider range of power and still be incorrect. 11/11/08

NRC K/A System/E/A **System** 045 Main Turbine Generator (MT/G) System

Number K5.17 **RO** 2.5* **SRO** 2.7* **CFR Link** (CFR: 41.5 / 45.7)

Knowledge of the operational implications of the following concepts as the apply to the MT/B System: Relationship between moderator temperature coefficient and boron concentration in RCS as T/G load increases

Question #: **64** | ***Question ID:* 8056807** | ☑ ***RO*** | ☐ ***SRO*** | ☐ Student Handout? | ☑ Lower Order?

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

Fuel is being moved in the Spent Fuel Pool (SFP) area during a refueling outage, when a SFP area radiation monitor fails high.

What affect would this radiation monitor failure have on the SFP ventilation system and can fuel handling operations in the SFP continue?

☐ **A** SFP ventilation will shift to AEAS mode, however Fuel Handling may continue.

☑ **B** SFP ventilation alignment will NOT be affected, Fuel tiandling may continue.

☐ **C** SFP ventilation can NOT shift to AEAS mode, therefore Fuel Handling must stop.

☐ **D** SFP ventilation has shift to EBFAS mode, therefore Fuel Handling must stop.

Justification

B - Correct; AEAS requires the triggering or failure of 2 area rad. monitors to actuate an AEAS (*214* logic).
A - Wrong; It is CTMT ventilation (Refuel Pool area) that requires only 1 rad monitor to actuate (114 logic), NOT SFP ventilation.
C - Wrong; Even with a rad monitor failed, any 2 of the other 3 can triggered and realign the SFP ventilation system to AEAS.
D -Wrong; An EBFAS actuating would block an AEAS and require fuel movement be stopped. However, only a manual actuation or SIAS can trigger an EBFAS.

References

LP ESA-01-C, ESAS Text Explanation of AEAS and actuation logic

NO Comments or Question Modification History at this time.

NRC K/A System/E/A System 072 Area Radiation Monitoring (ARM) System

Number K3.02 RO 3.1 SRO 3.5 CFR Link (CFR: 41.7 / 45.6)

Knowledge of the effect that a loss or malfunction of the ARM system will have on the following: Fuel handling operations

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

Question #: **65** | ***Question ID:*** *8680012* | ☑ ***RO*** | ▲ ***SRO*** | ☐ Student Handout? | ☐ Lower Order?

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

The plant is in "end-of-cycle" coastdown and workers have just begun erecting scaffolding in the "B" Emergency Diesel Generator (EDG) room. None of the plant systems have been tagged out yet.

Then, one of the scaffold bars hits a heat detector above the EDG, which causes the heat detector to fail in the "actuated" mode.

Which of the following describes the effect of this inadvertent actuation of this heat detector on plant systems or components?

☐ **A** The sprinkler system has NOT actuated because both a heat and a smoke detector actuation are required.

☐ **B** The 13U DG room deluge has been actuated, but the nozzles are NOT spraying down the EDG and the room.

☐ **C** The 13U Diesel Generator Deluge Supervisory Air System alarm activated on the Fire Panel in the control room.

☐ **D** The sprinkler system has NOT actuated because a second heat or smoke detector actuation is required.

Justification

B - Correct; Unlike the DC switchgear rooms, a single heat detector actuation will trigger the fire suppression system. However, the sprinkler system in the EDG rooms is a "dry" system. A fusible link in each nozzle must melt before that nozzle can spray down the room.
A - Wrong; This is how the Vital DC switchgear room fire suppression system works, NOT the EDG fire system. Although there are similarities, the two systems are different in actuation requirements.
C -Wrong; This would occur if the scaffolding broke a sprinkler head nozzle (the fusible link). The purpose of the supervisory air system is to detect the loss of a nozzle seal.
D - Wrong; Other suppression systems require a trigger on more than one detector to avoid an inadvertent actuation if just such an event occurs. However, because of the fusible links in each nozzle, the EDG system does NOT utilize that failure prevention method.

References

LP FPS-04-C, Pages 26, 28 & 29 Describe EDG Fire Deluge System.

NO Comments or Question Modification **History** at this time.

NRC K/A System/E/A System 086 Fire Protection System (FPS)

Number K6.04 RO 2.6 SRO 2.9 CFR Link (CFR: 41.7 / 45.7)

Knowledge of the effect of a loss or malfunction on the Fire Protection System following will have on the: Fire, smoke, and heat detectors

Question #: **66** | ***Question ID:*** *75144* | ☑ ***RO*** | ☐ ***SRO*** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The Letdown Valve Controller on Hot Shutdown Panel C-21 is shifted to MANUAL and given a maximum output signal.

Which one of the following describes the response that will be observed on the control room letdown flow controller on C02?

☐ **A** The controller output indicator will remain as-is, and letdown flow will not change from the value maintained by the C02 controller.

☑ **B** The controller output indicator will remain as-is, and letdown flow will go to the maximum allowed by a full open flow control valve.

☐ **C** The controller output indicator will track to 100%, and letdown flow will go to the maximum allowed by a full open flow control valve.

☐ **D** The controller output indicator will remain as-is, and letdown flow will go to the maximum flow allowed by the letdown limiter.

Justification

B - Correct; The C-21 controller is downstream in the control circuit from the C02 controllers and, therefore, have no effect on the C02 controller operation or output. However, the letdown valves will fully open, based on the output from the C21 controller because the C21 controller output does NOT go through the Letdown Limiter circuit.
A - Wrong; This is true of the controllers on C10 because those controllers first have to be put in the circuit by the "isolate" switch.
C - Wrong; The flow control valve operation is correct but the indication stated is based on that seen for Foxboro controllers that utilize the Foxboro IA system.
D - Wrong; Controller indication is correct but the letdown limiter is "upstream" of this controllers output and, therefore, has NO effect on the controllers output to the valves.

References

Letdown Flow Control Valve Signal Flow Path Diagram

Comments and Question Modification History

Added "control room" to question stem.

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

Generic KIA Selected

NRC KIA Generic **System** 2.1 Conduct of Operations

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

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

Question #: **67** | **Question ID: 54123** | ☑ **RO** ☐ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

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

While an operator is performing the independent verificatior; of a valve line-up, he discovers a valve is throttled 25% open instead of being in the required fully open position.

What action must be taken by the operator?

☐ **A** Open the valve fully and make a note in the comment section of the valve line-up checklist.

☐ **B** Direct the individual that originally positioned the valve to open the valve, then reperform the independent verification.

☑ **C** Contact the Shift Manager for guidance on the desired position before repositioning the valve.

☐ **D** Open the valve fully, then notify the Shift Manager that the position of the valve must be reverified.

Justification

C - CORRECT; Per administrative requirements, "If a discrepancy is discovered (the performers of a valve lineup) immediately notify first line supervision for resolution."
A - WRONG; NOT allowed even with permission because now only ONE check has been made of the valve.
B -WRONG; ONLY the "On Watch" US or SM can direct a mispositioned valve be repositioned.
D - WRONG; This would only be allowed if the operator had already been given instructions to re-align any mispositioned valves, such as restoring from a tagout.

References

OP-AA-5000, Independent Verification Requirements

NO Comments or Question Modification History at this time.

NRC KIA System/E/A **System** 2.1 Conduct of Operations

Generic K/A Selected

NRC KIA Generic **System** 2.1 Conduct of Operations

Number 2.1.29 **RO** 4.1 **SRO** 4.0 **CFR Link** (CFR: 41.10 145.1 145.12)

Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.

Question #: **68** **Question ID: 74001** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☑ **Lower Order?**

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

Which of the following decisions would be considered NON-CONSERVATIVE action?

☑ **A** Initiation of SRAS with the RWST level at 20% and dropping during a large-break LOCA.

☐ **B** Initiation of Once-Through-Cooling early due to one facility of equipment being unavailable due to loss of vital AC power.

☐ **C** Delaying watch relief past a 12 hour shift because the reactor is achieving its fifth (5th) doubling of counts on a startup.

☐ **D** Maintaining the plant at power (Mode 1) with a loss of all three Auxiliary Feedwater Pumps.

Justification

A - CORRECT; Early initiation of SRAS is non-conserve because of the loss of inventory to the CTMT sump
B - WRONG; EOPs state early initiation of OTC should be attempted if less than optimal equipment is available.
C - WRONG; reactor startup procedure states that shift turnover should NOT occur with the reactor approaching criticality.
D - WRONG; Although Tech Specs eventually require a plant shutdown if any Aux. Feed Pump is inoperable, loss of main feed post-trip would leave the plant without any source of SG feed flow.

References

OP-2260, Pg. 12, St. 1.8.2

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 2.1.39 **RO** 3.6 **SRO** 4.3 **CFR Link** (CFR: 41.10 143.5 145.12)

Knowledge of conservative decision making practices.

Question #: **69** | ***Question ID:*** ***64620*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ Student Handout? | ☑ Lower Order?

Rev. **4** | ☐ Selected for Exam | ***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** The alarm circuit is out of calibration and can not be solely utilized as an indication that mitigating actions are required.

B The monitored equipment associated with the annunciator is inoperable, but an alarm may indicate a status change.

☐ **C** The associated alarm logic card has been removed, preventing the annunciator from triggering by any input device.

☐ **D** The annunciator can be triggered by multiple inputs and one or more of these inputs are known to be out of service.

Justification

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.
A - Wrong; A trouble Report would be filled for this situation, containing details of the problem. An orange label (with magnetic backing), containing the TR number and date, would be placed on the control board as close to the alarm as possible
B - Wrong; This implies the equipment is out of service due to a "blue" tag, which would allow technicians to test the component and possibly cause a periodic alarm. However, an item out of service due to a tagout requires a RED dot, regardless of the tag color.
D -Wrong; This alarm condition would require an ORANGE dot on the window.

References

OP-2387A, Definitions

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 KIA Generic System 2.2 Equipment Control

Number 2.2.14 RO 3.9 SRO 4.3 CFR Link (CFR: 41.10143.3 145.13)

Knowledge of the process for controlling equipment configuration or status.

Question #: **70** | **Question ID: 80785** | ☐ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

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

During a refueling outage, the "A" TBCCW Pump breaker, B0107, is properly red tagged for troubleshooting due to an electrical problem. Electrical Maintenance has determined that the breaker has to be removed and bench tested in the maintenance shop. The intent is to install the original breaker when repairs are complete.

The tag on the breaker ____________

☐ **A** must be cleared before removal of the breaker from the cubicle.

☐ **B** must be lifted and held until the breaker is placed back in the cubicle.

☐ **C** remains with the breaker during the troubleshooting and repair process.

☑ **D** must be removed from the breaker and attached to the cubicle door

Justification

D is correct. OP-MP-200.1001, Equipment Clearance Process, Attachment 1, Section 3.1 describes the process for removing an MCC bucket from its cubicle when the breaker has been red tagged as follows: Remove tag, Remove MCC bucket from cubicle, Rehang tag on door.
A is incorrect. Even though the breaker is physically removed from the cubicle and the pump cannot be started, the tagging procedure does not allow for clearing the tag to remove the breaker.
B is incorrect. Even though the breaker is physically removed from the cubicle and the pump cannot be started, the tagging procedure does not allow for lifting the red tag until the breaker is reinstalled.
C is incorrect. Even though the breaker is physically removed from the cubicle and the pump cannot be started, the tagging procedure does not allow for keeping the tag with the breaker.

References

OP-MP-200-1001, Equipment Clearance Process, Attachment 1, Section 3.1

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 2.2.13 **RO** 4.1 **SRO 4.3** CFR **Link** (CFR: 41.10145.13)

Knowledge of the tagging and clearance procedures.

Question #: 71 | **Question ID:** 80625 | ☑ RO | ☐ SRO | ☑ Student Handout? | ☐ Lower Order?

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

The plant was operating at 100% power, steady state, with all CEAs fully withdrawn.

Then, CEA #1 in Reg. Group 7 drops to the bottom of the core (0 steps withdrawn)
The crew subsequently performs all required actions, per the applicable AOP, and is awaiting the I&C "go-ahead" to begin recovery of the dropped CEA.

It has now been one hour and 50 minutes since the CEA dropped, and I&C has just informed the control room that CEA recovery steps may begin.

Which of the following actions must be performed, based on the existing conditions?

☑ **A** Do NOT perform CEA recovery steps. Immediately commence a plant shutdown to MODE 3.

☐ **B** Ensure the dropped CEA is withdrawn to at least 170 steps within the next 10 minutes.

☐ **C** Enter LCO 3.0.3. and withdraw the CEA to at least 170 steps within the next 60 minutes.

☐ **D** Trip the plant and maintain the reactor shut down for a minimum of 2 hours before restarting.

Justification

A - Correct; Per the requirements of TSAS 3.1.3.1 action A.I and AOP 2556 step 4.28.k: IF the misaligned CEA is not realigned to within 10 steps of all other CEAs in its' group within 2 hours, PLACE the plant in HOT STANDBY condition within the next 6 hours.
B - Wrong; This fast a CEA recovery is in violation of the AOP guidance, as it has a strong possibility of damaging the fuel. Based on the AOP recovery guidelines, it is mathematically impossible to recover the dropped CEA within the Tech. Spec. time limit.
C - Wrong; Although the Tech. Spec. time limit for CEA recovery can NOT be met, entry into T.S.A.S. 3.0.3 does NOT allow continued recovery of the dropped CEA. The changing distribution of xenon in the surrounding fuel puts the core in an unanalyzed condition.
D - Wrong, A plant trip is NOT required and would be considered an overly aggressive plant shutdown and a non-conservative action.

References | **Provided**

Tech. Spec. 3.1.3.1[provided to Examinees]; CEA Position, Action " A ; Misaligned by >20 Steps.
AOP-2556 [NOT provided to Examinees]; Step 4.28, Misaligned CEA Recovery.

(NOComments or Question Modification History at this time.

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

Generic KIA Selected

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

Number 2.2.40 **RO** 3.4 **SRO** 4.7 **CFR Link** (CFR: 41 10 143.2 143.5 145.3)

Ability to apply Technical Specifications for a system.

Question #: 72 | **Question ID: 8500016** | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

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

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 beer granted per the Emergency Exposure Limits guidelines.

Which of the following exposure requirements are still applicable for the volunteer who enters the high radiation area to assist the injured operator?

☐ **A** Any male or 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.

Justification

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.
A - Wrong; This equates to a dose of 5 rem, which is the normal limit for non-emergency scenarios.
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.
C - Wrong; This is the correct dose for life-threatening emergency situations. However, although excluding females is plausible, it is NOT an administrative requirement.

References

MP-26-EPI-FAPO9; Radiation Exposure Controls

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: 41.12 143.4 / 45.10)

Knowledge of radiation exposure limits under normal or emergency conditions.

Question #: **73** **Question ID:** **1000109** ☑ **RO** ☐ **SRO** ☐ Student Handout? ☑ Lower Order?

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

Radiography is being performed in the Auxiliary Building and has caused an area radiation monitor to alarm.

The US has directed the applicable module on RC-14 be placed in ALARM DEFEAT until the operation is complete.

Which of the following describes why the radiation monitor's switch is placed in the ALARM DEFEAT position?

☐ **A** To silence the radiation monitor's horn on the local module.

☐ **B** To reset any automatic action caused by the radiation monitor.

☐ **C** To clear the radiation monitor's red and/or amber lights on RC-14.

☑ **D** To allow other radiation monitor alarms to annunciate on C-06/7

Justification

D - Correct; Placing the applicable ALARM DEFEAT switch in he ALARM DEFEAT position will allow other area radiation monitor alarms to be annunciated on C-06/7. The red 'HIGH' and amber "'FAIL' lights will be lit on the applicable rad monitor on RC-14. The local horn will need to be bypassed with a key on the local module.
A - Wrong; The ALARM DEFEAT switch will NOT silence the local horn.
B -Wrong; The ALARM DEFEAT switch will NOT reset any automatic action caused by the rad monitor. In fact, the ALARM DEFEAT switch will result in a rad monitor failure which will prevent resetting any automatic function.
C - Wrong; The ALARM DEFEAT switch will NOT clear the red and amber lights on the RC-14 module. In fact, the ALARM DEFEAT switch will cause the red and amber lights to be lit.

References

ARP-2590E-128, Defeat Area Rad Monitor Alarm

(NOComments or Question Modification History at this time.

NRC K/A System/E/A System 2.3 Radiation Control

Generic KJA Selected

NRC KIA Generic System 2.3 Radiation Control

Number 2.3.15 RO 2.9 SRO 3.1 CFR Link (CFR: 41.12143.4 / 45.9)

Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Question #: **74** | **Question ID: 8310054** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

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

Then, a seismic event occurs resulting in the following plant conditions:

- Pressurizer level and pressure are dropping rapidly.
- Containment pressure is <1 psi and stable.
- RBCCW surge tank is rising rapidly.
- 'B' RCP has tripped and 'A' RCP has been manually secured.

Which of the following describes actions required for this event?

☑ **A** Align 'A' HPSI pump and start it, then secure 'B' LPSI pump, realign LPSI pumps for Safety Injection, then manually start both LPSI pumps.

☐ **B** Ensure the 'A' HPSI and 'A' LPSI Pumps have automatically started and their associated injection valves automatically have opened.

☐ **C** Take 'A' HPSI Pump handswitch to OFF, secure SDC and realign LPSI pump suction, and observe automatic start of Facility 1 SI pumps.

☐ **D** Manually actuate SIAS via C01 push button and verify both Facility 1 SI Pumps automatically start and all safety injection valves realign.

Justification

A - Correct; SIAS is blocked and will NOT respond on lowering pressurizer pressure. Accordingly, the 'A' HPSI is required to be in PTL due to lowered RCS temperature and LPSI alignment prevents injection until realigned, these actions are directed by OP 2207 in response to a LOCA.
B -Wrong; This would occur if Facility 1 HPSI and LPSI were maintained fully operable and SIAS was NOT blocked or the LOCA was raising CTMT pressure.
C - Wrong; This would occur if LPSI injection valves were powered and SIAS was NOT blocked or the LOCA was raising CTMT pressure.
D -Wrong; This would occur if LPSI did NOT needs to be manually realigned and the HPSI pump "Pull-To-Lock" feature allowed for automatic start on an ESAS actuation (similar to the Charging Pumps).

References

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

NO Comments or Question Modification History at this time.

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

Generic KIA Selected

NRC KIA Generic **System** 2.4 Emergency Procedures /Plan

Number 2.4.9 **RO** 3.8 **SRO** 4.2 **CFR Link** (CFR: 411 0 143.5 145.13)
Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Question #: 75 | Question ID: 8000051 | a RO | a SRO | ☐ Student Handout? | a Lower Order?
Rev. 0 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

An RO qualified operator, attached to Crew "D", is on site, temporarily assigned to the Procedure Group.

Then, Crew "E" operators, who presently have the watch in the control room, declare an emergency event requiring activation of the Station Emergency Response Organization (SERO). The Unit 2 Shift Manager (SM) makes an announcement over the site paging system, for all SERO members to report to their designated emergency response facility.

Which of the following describes the location where the Crew "D" RO must now report, based on the SM's site page?

☐ **A** The Unit 2 Control Room.

☐ **B** Remain in Work Control.

☑ **C** Building 475 Cafeteria.

☐ **D** Technical Support Center.

Justification

C - CORRECT; All "OFF-Shift" Operations personnel are required to report to the Operational Support Center Assembly Area, which is the cafeteria in building 475. They will be dispersed from there to various plant locations, as they are needed.
A - WRONG; The RO temporarily assigned to the Procedure Group is NOT considered an "On-Shift" operator, which means he should NOT go to the Control Room. Only operators that actually have the "Shift" are supposed to report to the Control Room during an event when SERO is activated.
B - WRONG; This would only be correct if the announcement stated personnel should seek immediate shelter, such as during a security event or fast moving natural phenomenon.
D - WRONG: Some support personnel with SERO positions, temporarily assigned to the Procedure Group, would report to this location. But Not an RO qualified operator.

References

SERO CBT, Slide 20, Operator SERO Reporting Requirement

(NOComments or Question Modification History at this time.

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

Generic KIA Selected

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

Number 2.4.29 **RO** 3.1 **SRO** 4.4 CFR **Link** (CFR: 41.10143.5 145.11)
Knowledge of the emergency plan.

Question#: **76** | ***Question ID:* 8000025** | ☐ **RO** | ☑ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The plant was tripped from 100% power due to a circulating water header rupture in the East condenser pit which resulted in the condenser pit water level reaching 10 inches. All equipment responds as expected. The Standard Post Trip Actions of EOP 2525 have been completed and the crew has just entered EOP 2526, Reactor trip Recovery.

Suddenly, the RSST is lost.

The following plant conditions now exist:

- Pressurizer pressure — 2060 psia and going up slowly
- Pressurizer Level — 40% and rising
- SG Pressures — 890 psia and stable
- SG levels — 55% in both *SG* and steady
- 'A' and 'B' DGs — Supplying associated vital buses
- Tcold — 530 °F and stable
- Thot — 536 °F and rising

All other equipment and parameters are as expected.

Which of the following actions must now be directed by the US?

☐ **A** Perform EOP 2525, Standard Post Trip Actions, in it's entirety to ensure all Safety Functions are met.

☑ **B** Return to the beginning of EOP 2526, Reactor Trip Recovery, and review Appendix 1, Diagnostic Flowchart.

☐ **C** Go To AOP 2583, Loss of All AC During Shutdown Conditions, and AOP 2517, Circulating Water Malfunctions.

☐ **D** Immediately transition to EOP 2540, Functional Recovery, as a result of the two events that are in progress.

Justification

B is correct. OP 2260, Rev 009-02, step 1.9.4c states, "If, during the performance of an ORP, a major change in plant conditions occurs, the US should return to the beginning of the procedure in use and commence the procedure again at the Confirm Diagnosis section, which, if appropriate, will lead to review of Appendix 1, Diagnostic Flowchart." Therefore, the US is required to return to step 1 and review the Diagnostic Flowchart to determine the appropriate procedure for the given conditions.
A is incorrect. If a major change in plant conditions occurs during the performance of EOP 2525, then the US should return to the beginning and commence the procedure again. In this case, EOP 2525 is complete.
C is incorrect. Although the listed AOP would provide guidance to mitigate these two events, OP 2260 does not direct the US to enter any AOPs directly from an EOP.
D is incorrect. From an EOP perspective, only one emergency event is in progress, the loss of the RSST. The other event, the loss of Circulating Water, is simply an abnormal event and does NOT require a transition to EOP 2540.

References

OP 2260, Rev 009-02, step 1.9.4c

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** E02 Reactor Trip Recovery

Number EA2.1 **RO** 2.7 **SRO** 3.7 **CFR Link** (CFR: **43.5** 145.13)

Ability to determine and interpret facility conditions and selection of appropriate procedures during abnormal and emergency operations as they apply to the Reactor Trip Recovery.

Question #: 77 | **Question ID: 8000036** | ☐ **RO** | ☑ **SRO** | ☑ Student Handout? | ☐ Lower Order?

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

The plant was at 100% power when a Loss of Load caused an automatic Reactor trip. The US has directed entry into EOP 2525, Standard Post trip Actions.

During the performance of EOP 2525, Standard Post Trip Actions, the following conditions are reported:
- Pressurizer level is 100%.
- RCS pressure is 1420 psia and lowering.
- RCS temperature is 529°F and stable.
- 2-RC-402, " A PORV, discharge temperature is approximately 320°F and stable (only red position indication light is lit)
- 2-RC-404, "B" PORV, discharge temperature is approximately 135 and slowly lowering (only green position indication light is lit)
- Pressurizer Safety Valve discharge temperatures are approximately 106°F and stable
- Quench Tank pressure is 90 psig and lowering.
- Quench Tank level is 98% and stable.
- Containment pressure is 6 psig and rising
- SIAS, CIAS, and EBFAS have successfully initiated

The crew is performing (or will perform) all the required actions for the above conditions

Assume appropriate notifications and classifications will be made by the Shift Manager.

The crew has been directed to proceed to COLD SHUTDOWN for repairs?

As a spare SRO on shift, what additional administrative actions must be taken?

☑ **A** Restore the PORV to OPERABLE status within one hour or deenergize the associated PORV Block Valve after it's closed.
Restore the PORV to OPERABLE status within 24 hours of RCS temperature dropping below 200°F or depressurize and vent the RCS within the next 8 hours.

☐ **B** Restore the PORV to OPERABLE status within one hour and maintain power to the associated PORV Block Valve after it's closed.
Restore the PORV to OPERABLE status within 7 days of RCS temperature dropping below 300°F or depressurize and vent the RCS within the next 8 hours.

☐ **C** Restore the PORV to OPERABLE status within one hour or deenergize the associated PORV Block Valve after it's closed.
Restore the PORV to OPERABLE status within 72 hours of shutting down the reactor and main turbine or cooldown the RCS to <300°F within the next 6 hours.

☐ **D** Restore the PORV to OPERABLE status within one hour and maintain power to the associated PORV Block Valve after it's closed.
The PORV block valve must be maintained closed and de-energized until the RCS is <300°F or cooldown the RCS to <300°F within 6 hours and <200°F within the next 6 hours.

Justification

A is correct. The conditions presented are indicative of a stuck open PORV. Technical Specification LCO 3.4.3, ACTION b. requires the PORV Block Valve associated with the inoperable PORV to be closed within 1 hour, with power maintained. Additionally, Technical Specification LCO 3.4.9.3, requires that, with only one OPERABLE PORV, within 24 hours of reaching MODE 5 the PORV must be restored to OPERABLE or the RCS must be depressurized and vented within the next 8 hours.
B is incorrect. The affected PORV cannot be cycled; therefore, power must be removed from the associated block valve once it is closed. Additionally, Technical Specification LCO 3.4.9.3, requires that, with only one OPERABLE PORV, within 24 hours NOT 7 days as is the case for MODE4) of reaching MODE 5 the PORV must be restored to OPERABLE or the RCS must be depressurized and vented within the next 8 hours.
C is incorrect. This distracter does not consider Tech Spec 3.4.9.3, which becomes applicable when entering MODE 4 or MODE 5.
D is incorrect. The affected PORV cannot be cycled; therefore, power must be removed from the associated block valve once it is closed. Additionally, This distracter does not consider Tech Spec 3.4.9.3, which becomes applicable when entering MODE 4 or MODE 5.

References | **Provided**

T.S. 3.4.9.3, Action b
T.S. 3.4.3
Selected Unit 2 Technical Specifications without Bases.

Question #: 78 | **Question ID:** 8000027 | ☐ RO ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

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

The plant is in Natural Circulation after a reactor trip from 100% due to a small break LOCA with a concurrent loss of the RSST. Standard Post Trip Actions, EOP 2525, has been successfully performed. You have directed the RO to check for single phase natural circulation in accordance with Loss of Coolant Accident, EOP 2532.

Which of the following sets of stable parameters is acceptable (administratively allowed) for Natural Circulation after the cooldown and depressurization has been initiated?
(Assume the cooldown limitations of the Technical Specifications and the Technical Requirements Manual are maintained.)

☐ **A** Pressurizer pressure is 812 psia
Pressurizer level is 25%
Highest CET is 520°F
Tc is 485°F
Th is 515°F

☐ **B** Pressurizer pressure is 1100 psia
Pressurizer level is 12%
Highest CET is 520°F
Tc is 455°F
Th is 518°F

☐ **C** Pressurizer pressure is 610 psia
Pressurizer level is 27%
Highest CET is 455°F
Tc is 423°F
Th is 443°F

☑ **D** Pressurizer pressure is 931 psia
Pressurizer level is 8%
Highest CET is 505°F
Tc is 465°F
Th is 500°F

Justification

D is correct. Loop delta-T is 35°F (maximum is 55"); Th and Tc are constant (stated in question stem); CET subcooling is 31°F (Minimum operating limit of the PIT curve for CETs is 30°F); Difference between Th and CET temperature is 5°F (maximum is 10°F). Pressurizer level of greater than 20% is NOT a requirement for natural circulation during a LOCA (Minimum, of 20% for all other events). 8% correlates to the minimum PZR level displayed by the PPC, regardless of how low PZR level then drops.
A is incorrect. A CET temperature of 512°F is the saturation temperature for 812 psia. Although Tc is greater than 30°F subcooled, this does not meet the requirement for natural circulation.
B is incorrect. Although a CET temperature is 36°F subcooled for 1100 psia (meets the requirement of greater than 30°F), the loop delta-T of 63°F does not meet the requirement of less than 55°F for natural circulation.
C is incorrect. Although a CET temperature of 455°F is approximately 33°F subcooled, Th and CET temperature are greater than 10°F, which does not meet the requirements for natural circulation.

References

EOP-2532, St. 39, Verification of Single Phase NC Flow

Comments and Question Modification History

Although all reviewers missed this question, it was determined to be valid based on later discussions. This may indicate a knowledge deficiency issue and will be discussed with supervision after the exam is administered. 11/14/08

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

Number EA2.37 RO 4.2 SRO 4.5 CFR Link (CFR 43.5 145.13)

Ability to determine or interpret the following as they apply to a small break LOCA: Existence of adequate natural circulation

Question #: 79 **Question ID: 54119** ☐ RO ☑ SRO ☐ Student Handout? ☐ Lower Order?

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

The following conditions exist:
- A Large Break LOCA occurred approximately 9 hours ago.
- SRAS was initiated approximately 7.5 hours ago.
- Chemistry Department reports the current RCS sample indicates 1500 ppm boron concentration.
- An RCS sample taken 4 hours ago indicated a boron concentration of 1945 ppm.
- CET temperature is 374°F.
- Both Facility 1 and Facility 2 are available.
- All other conditions are as expected for this event.

Given these conditions, which one of the following is the procedurally preferred operator action that must be directed by the US?

☐ **A** Stop the 'A' HPSI Pump, ensure 'C' HPSI pump is running, and align the 'A' HPSI pump to auxiliary spray.

☐ **B** Stop the 'A' LPSI Pump, ensure 'B' HPSI pump is running, and align the 'A' LPSI pump to auxiliary spray.

☐ **C** Make-up to the BASTs and restart all of the available charging pumps injecting to the 'A' HPSI header.

☐ **D** Align the 'B' LPSI pump to inject to the RCS through the SDC suction line, maintain HPSI injecting as-IS.

Justification

A - CORRECT; Boron precipitation is expected. With both facilities available and CETs greater than 345°F, a Facility 1 HPSI pump is utilized to establish flow through the PZR spray nozzle (via Aux. spray), through the Pressurizer into the #2 hot leg, and back through the core such that water in the core is flushed out the cold leg break. Flow via this path is required to prevent the boric acid concentration in the fuel region from reaching the level at which crystallization would occur. (EOP 2541, App. 18-B)
B - WRONG; Based on given conditions, both LPSI pumps are secured.
C -WRONG; This would be the actions if additional boron injection were required and the normal path was not available.
D -WRONG; This is a partial alignment for the alternate Boron precipitation control path.

References

EOP-2532, St. 56, Hot Leg Injection.
EOP-2541, App. 18, Hot Leg Injection w/ both Fac. Available

NO Comments or Question Modification History at this time.

NRC KIA System/E/A System 011 Large Break LOCA

Number EA2.11 RO 3.9 SRO 4.3 CFR Link (CFR 43.5 145.13)

Ability to determine or interpret the following as they apply to a Large Break LOCA: Conditions for throttling or stopping HPI

Question #: **80** | *Question ID:* 8000029 | ☐ RO | ☑ SRO | ☑ Student Handout? | ☐ Lower Order?

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

The plant is operating at 100% power when Pressurizer Pressure Safety Channel " A transmitter fails HIGH. Actual Pressurizer pressure is confirmed to be 2250 psia and stable.

In addition to submitting a Condition Report, as the US, which of the following is the MINIMUM action that you must direct the Reactor Operator to take?

☐ **A** Within one hour, place both Automatic Auxiliary Feedwater Override/Man/Start/Reset switches in the Pull-To-Lock position, obtain the appropriate key from the key locker, and bypass ATWS on Channel " A Panel C-100.

☐ **B** Within 48 hours, place TM/LP Trip on Channel " A RPS, High Pressurizer Trip on Channel " A RPS, Pressurizer Pressure Bistable on Channel " A ESAS, and ATWS on Channel " A Panel C-100, in a tripped condition.

☐ **C** Within one hour, obtain the appropriate keys from the key locker and bypass TMLP Trip on Channel " A RPS, High Pressurizer Trip on Channel " A RPS, Pressurizer Pressure Bistable on Channel " A ESAS, and ATWS on Channel " A Panel C-100.

☐ **D** Within 48 hours, place both Automatic Auxiliary Feedwater Override/Man/Start/Reset switches in the Pull-To-Lock position; place High Pressurizer Trip Unit on Channel " A RPS and ATWS Trip Unit on Channel " A Panel C-100 in a tripped condition.

Justification

C is correct. Technical Specification 3.3.1.1, Table 3.3-1, Action 2 requires placing the inoperable channel and all affected functional units that receive an input from the inoperable channel in either a bypassed or a tripped condition. However, the Tech. Specs. do NOT state specifically what items receive an input. With Pressurizer Pressure Safety Channel " A transmitter failed high, TMLP trip on RPS, High Pressure Trip on RPS, Low Pressure SIAS on ESAS, and ATWS on C-100 are affected functional units that receive an input from the inoperable channel and must be bypassed or placed in a tripped condition within one hour per ARP 2590C-017.
A is incorrect. The Automatic Auxiliary Feedwater Override/Man/Start/Reset switches are NOT required to placed in the Pull-To-Lock position. On an actual 2 out of 4 high Pressurizer pressure, Auxiliary Feedwater would automatically start if pressure remained high for a prescribed period of time. Placing both Automatic Auxiliary Feedwater Override/Man/Start/Reset switches in the Pull-To-Lock position will prevent the automatic start. Disabling the automatic start is NOT appropriate for this condition
B is incorrect. If Pressurizer Pressure Safety Channel " A transmitter is not restored to OPERABLE status within 48 hours, then the inoperable channel (and all related functional units must be placed in the tripped condition; otherwise, bypassing the channel within one hour is appropriate.
D is incorrect. See distracters A and B.

References **Provided**

Tech. Spec. 3.3.1.1 and Table 3.3-1 and Action 2.
Handout: Selected Unit 2 Technical Specifications without Bases.

NO Comments or Question Modification History at this time.

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

Generic KIA Selected

NRC WA Generic **System** 2.1 Conduct of Operations

Number 2.1.30 **RO** 3.9 **SRO** 3.4 **CFR Link** (CFR: 41.7 145.7)

Ability to locate and operate components, including local controls.

Question #: **81** | **Question ID: 8000030** | ☐ **RO** | ☑ **SRO** | ☐ Student Handout? | ☑ Lower Order?

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

Millstone Unit 2 is designated an eight hour coping plant for a Station Blackout event.

What component or system function provides the basis for the time limitation?

☐ **A** If ventilation is restored to the vital systems within one hour of a Station Blackout, then components will NOT overheat and the plant may remain in MODE 3 for an additional 7 hours.

☐ **B** With the Condensate Storage Tank at the minimum Tech Spec level, the plant may remain in MODE 3 for at least 8 hours following a Station Blackout.

☐ **C** With a maximum Tech Spec RCS leakage of 25 gpm from RCP Bleedoff, the core will remain covered for at least 8 hours following a Station Blackout.

☐ **D** If one Charging Pump is started within the first hour of a Station Blackout, then, with maximum assumed RCS leakage, the core will remain covered for an additional 7 hours.

Justification

D is correct. In a Station Blackout, power is assumed to be restored to at least one vital bus within one hour to allow starting various loads, including a charging pump. With the maximum allowed Tech Spec RCS leakage (12 gpm) coupled with the assumed maximum RCP Bleedoff leakage of 100 gpm (25 gpm per RCP), the maximum assumed leakage of 112 gpm will NOT uncover the core for at least 7 more hours.

A is incorrect. If ventilation is NOT restored with 30 minutes, action must be taken to prevent damage to vital components from overheating. There is NO mode dependency on the restoration of ventilation.

B is incorrect. Minimum Tech Spec level in the Condensate Storage Tank is adequate to maintain HOT STANDBY conditions for at least 10 hours after a loss of Offsite Power (NOT Station Blackout).

C is incorrect. Maximum Tech Spec allowed leakage is approximately 12 gprn. RCP Bleedoff is NOT used to determine RCS leakage.

References

LP E30-01-C, R-3, Pg. 9, Assumed power restoration criteria

Comments and Question Modification History

Although this question was missed by 3 out of 4 reviewers, it was later determined to be valid. This may indicate a knowledge deficiency issue and will be discussed with supervision after the exam is administered. 11/14/08

NRC K/A System/E/A System 055 Station Blackout

Generic KIA Selected

NRC **KIA Generic** System 2.1 Conduct of Operations

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

Knowledge of the purpose and function of major system components and controls, Question #: 82 | **Question ID: 8000032** ☐ RO ☑ SRO ☐ Student Handout? ☑ Lower Order?

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

The crew is shutting down the plant for a refueling outage in accordance with OP 2205, Plant Shutdown and OP 2206, Reactor Shutdown.

Shortly after the Reactor Trip Breakers are opened, a Process Radiation Monitor Hi-Hi/Fail annunciator alarms immediately followed by an automatic isolation of SG Blowdown. Pressurizer pressure and level are verified to be stable at their present values. The crew requests a Chemistry sample on both Steam Generators.

Chemistry reports the following:
- #1 S/G frisk indicates 120 counts per minute above background
- #2 S/G frisk indicates background

Which of the following subsequent procedures should the US utilize to mitigate the event and continue the plant cooldown?

☑ **A** AOP 2569, Steam Generator Tube Leak, to isolate #1 Steam Generator. Then, continue the cooldown utilizing OP 2207, Plant Cooldown.

☐ **B** EOP 2534, Steam Generator Tube Rupture, to isolate #1 Steam Generator. Then, transition to EOP 2541, Appendix 8, Plant Cooldown.

☐ **C** AOP 2569, Steam Generator Tube Leak, to isolate #1 Steam Generator. Then, transition to EOP 2541, Appendix 8, Plant Cooldown.

☐ **D** EOP 2534, Steam Generator Tube Rupture, to isolate #1 Steam Generator, Then, continue the cooldown utilizing OP 2207, Plant Cooldown.

Justification

A is correct. AOP 2569 contains the required steps to isolate the most affected Steam Generator. When this is accomplished, the AOP directs the crew to OP 2207 to complete the cooldown.
B is incorrect. It would be inappropriate to enter EOP 2534 to isolate the affected S/G because the conditions are indicative of only a tube leak, NOT a tube rupture. Additionally, EOP 2541, Appendix 8 would NOT be used to complete the cooldown to SDC because the procedure does NOT provide guidance on concurrent RCP operation when initiating SDC. (Concurrent RCP operations are always preferred in non EOP events.)
C is not correct. The crew would NOT be directed to EOP 2541, Appendix 8, from EOP 2534. (Also, see justification for B. above.)
D is incorrect. Although EOP 2534 provides the guidance to isolate the most affected SIG, there is NO procedural guidance for the transition between EOP 2534 and OP 2207. AOP 2569 provides the guidance to cool down to a Th of 515°F and isolate the affected S/G.

References

AOP-2569; Steam Generator Tube Leak, App. Pages

NO Comments or Question **Modification History at this time.**

NRC K/A System/E/A System 037 Steam Generator (SIG) Tube Leak

Generic K/A Selected

NRC KIA Generic System 2.4 Emergency Procedures /Plan

Number 2.4.9 RO 3.8 SRO 4.2 CFR Link (CFR: 41.10 143.5 145.13)

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

Question #: **83** **Question ID:** 8000044 ☐ RO ☑ SRO ☐ student Handout? ☑ Lower Order?

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

The Rad. Waste PEO has just brought an Aerated Radioactive Waste (ARW) Monitor Tank discharge permit to the Shift Manager for review and approval.
Upon reviewing the permit and ARW system status, the SM has noticed that the ARW monitor tank was sampled by chemistry for the generation of the discharge permit with a level of 85%. However, the tank now has an actual level of 95%.

Which of the following describes the actions the Shift Manager is required to direct prior to discharging the applicable ARW Monitor Tank?

☐ **A** Re-calculate the allowable discharge flow rate based on the new, higher tank level, and annotate this on the existing discharge permit.

☑ **B** Re-mix the tank for the required period of time, then resample the tank and generate a new discharge permit based on the new sample.

☐ **C** Re-sample the tank and generate a second discharge permit and discharge the tank based on the most conservative of the two permits.

☐ **D** Re-mix the tank contents to ensure thorough mixing with the previously sampled contents and discharge the tank on the existing permit.

Justification

B - CORRECT: Althouah the calculated discharge flow rate is one of the items the SM must verifv when approving the permit for release, ARW discharges are required to be done in "batch" mode. This means the tank contents must be a discrete quantity with a known level of radioactivity. Once the tank showed signs of additional input, the contents were no longer known. Therefore, the tank must be re-sampled before it could be legally discharged.
A - WRONG; This is actually what is done for every discharge to ensure the actual amount of the discharge is correctly documented. This must be done because the total amount listed on the permit assumes every gallon of tank volume will be discharged. However, the discharge pump can NOT pump the tank down to zero and often trips off line with several percent left in the tank.
C - WRONG; This is what is done if the ARW discharge Rad. Monitor is NOT operable and the tank must still be discharged based on plant needs. However, administrative requirements state that if the Rad. Monitor is operable it will be used per existing guidelines.
D -WRONG; This is acceptable if the contents had been known (by sampling) but had stratified and was not indicating properly on the Discharge Rad. Monitor.

References

SP-2617A, Pg 6, Precaution 3.2
REMODCM, Rad Waste Sampling Requirements (Batch Discharge).

Comments and Question Modification History

Difficult K/A. There are NO radiation monitors associated with Technical Specifications, although several are referenced in the REMODCM. Is this fair game? 11/14/08

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

Generic KIA ~elected],

NRC WA Generic **System** **2.2** Equipment Control

Number 2.2.40 **RO** 3.4 **SRO** 4.7 **CFR Link** (CFR: 41.10 / 43.2 143.5 / 45.3)

Ability to apply Technical Specifications for a system.

Question #: **84** | ***Question ID:*** ***8153383*** ☐ ***RO*** ✔ ***SRO*** ☐ Student Handout? ☐ Lower Order?

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

The Outside Rounds PEO reports that a tractor trailer truck has just accidentally crashed into the North Fire Water Tank. The tank has a very large hole in it and is rapidly draining. The South Fire Water Tank is NOT affected. The US has directed the supply valve to the North Fire Water Tank to be closed. Unit 3 reports that the tank is draining at a rapid rate and is expected to be empty within the next few minutes.

Which of the following statements describes the administrative concern with the Fire Water System?

☐ **A** If a hose station or sprinkler system were to be placed in service, the three Aux Feedwater Pumps become inoperable due to the loss the emergency source of makeup water. Immediately initiate action to restore at least one auxiliary feed pump to OPERABLE status.

☑ **B** Although the Fire Suppression System is still available for subsequent fire fighting activities, it must be considered inoperable. Restore the inoperable fire tank to OPERABLE status within 7 days or, develop a plan and schedule for the delivery of fire water.

C As long as at least one Fire Water Tank and an offsite water source are OPERABLE for subsequent fire fighting activities, there are NO administrative concerns for Unit 2.

☐ **D** The Fire Protection Suppression is available for fire fighting activities; however, all sprinkler and hose stations are inoperable. Within one hour, initiate a plant shutdown to MODE 5

Justification

B is correct. One fire water tank will still be available to fight fires; however, TRM 3.7.9.1, ACTION a. requires both tanks, each with a minimum volume of 200,000 gallons, for the fire protection system to be OPERABLE.
A is incorrect The Fire System condition or usage is NOT considered when determining the OPERABILITY of the Aux Feed System even though it is the emergency source of water for feeding S/Gs. The emergency source is NOT considered in Technical Specification LCO 3.7.1.2.
C is incorrect. With one Fire Water Tank less than 200,000 gallons, the Fire System is INOPERABLE regardless of the availability of an offsite makeup source.
D is incorrect. The hose stations and sprinkler system are still considered OPERABLE, even though one Fire Water Tank is NOT available as a source; therefore, Tech Spec LCO 3.0.3 is NOT applicable.

References

TRM 3.7.9.1, Fire Protection Systems

Comments and Question Modification History

Changed distractors to remove reference to specific Tech Spec or TRM numbers. This was considered as possibly 'bordering on direct look up'.

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

Generic **K/A Selected**

NRC KIA Generic System 2.2 Equipment Control

Number 2.2.37 RO 3.6 SRO 4.6 CFR Link (CFR: 41.7 143.5 145.12)

Ability to to determine operability and lor availability of safety related equipment.

Question #: **85** | *Question ID:* **8000060** | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

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

The plant automatically tripped due to a Steam Generator Tube Rupture on #1 Steam Generator with a subsequent loss of Offsite Power. The crew successfully completed EOP 2525, Standard Post Trip Actions. The affected Steam Generator has been isolated per EOP 2534, Steam Generator Tube Rupture. The subsequent cooldown (after lowering both hot leg temperatures to less than 515°F) has been continuing for the past hour at approximately 65°F/hr.

The board operators report the following conditions:

- Loop 1 Th is 467°F and lowering
- Loop 2 Th is 455°F and lowering
- Loop 1 Tc is 468°F and lowering
- Loop 2 Tc is 425° and lowering
- #1 S/G pressure is 456 psia and lowering
- #2 S/G pressure is 422 psia and lowering
- RCS pressure is 700 psia and lowering

Based on the above information, which of the following actions is procedurally appropriate to ensure adequate core cooling?

☐ **A** Raise the cooldown rate to between 80 and 100°F/hr.

☐ **B** Lower RCS pressure to between 550 and 600 psia.

☐ **C** Lower the cooldown rate to between 10 and 25°F/hr.

☐ **D** Raise RCS pressure to between 850 and 900 psia.

Justification

C is correct. A difference of more than 10°F in loop hot leg temperatures is indication of the S/Gs becoming 'uncoupled'. As a result, the isolated SIG becomes a heat source for the RCS and he cooldown begins to stall (i.e., core heat removal is NOT adequate). There are several methods that may be utilized to restore adequate cooling to both loops. The method used here is to slow the cooldown and allow the isolated SIG to equalize with the intact SIG.
A is incorrect. Raising the cooldown rate will cool and depressurize the intact SIG; however the isolated SIG will NOT cool down and will prevent depressurizing the RCS.
B is incorrect. Lowering RCS pressure will allow more safety injection flow, but will also lower subcooling below the low limit of 30°F. This will bring the RCS closer to boiling which may have a negative impact on core heat removal.
D is incorrect. Raising RCS pressure will NOT improve the cooldown on the affected SIG and will only cause the leakage form the RCS to the affected SIG to rise possibly causing the steam lines to flood.

References

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

Comments and Question Modification History

New Question as of 11110108.

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

Number EA2.04 **RO** 3.7 **SRO** 4.2 **CFR Link** (CFR 43.5 145.13)

Ability to determine or interpret the following as they apply to a Inadequate Core Cooling: Relationship between RCS temperature and main steam pressure

Question #: **86** | ***Question ID:*** ***8000047*** | ☐ ***RO*** | ☐ ***SRO*** | ☐ Student Handout? | ☑ **Lower** Order?

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

The PZR LVL BYPASS switch on Panel C-02 has a lamacoid (plastic label) by it with the following Caution: USE ONLY WHEN SUBCRITICAL.

Which of the following states the function of this switch when placed in the BYPASS position and the specific condition that would allow the US to direct placing it in the BYPASS position?

☐ **A** The automatic start of both backup Charging Pumps is disabled.
After a Reactor trip to allow manual control of Charging Pumps.

☐ **B** The upper and lower Letdown flow limitations are bypassed.
Prior to a plant heat up to allow starting Charging and Letdown.

☐ **C** The high level trip of both backup Charging Pumps is disabled.
During a plant cool down to maintain 40% Pressurizer level.

☐ **D** The low level trip of the Pressurizer heaters is bypassed.
Placing SDC in service while maintaining Pressurizer pressure.

Justification

C is correct. In order to allow raising Pressurizer level above the automatic high level trip of the backup Charging Pumps, the PZR LVL BYPASS switch must be placed in the Bypass position. This is allowed to be performed only when subcritical to ensure level is maintained within the Tech Spec limits and to prevent inadvertently filling the Pressurizer during power operation.
A is incorrect. The automatic start of both Charging pumps is NOT disabled with this switch. The automatic start on low level is defeated by placing the Charging pumps in Pull-To-Lock. This is how Pressurizer level is controlled post-trip.
B is incorrect. The PZR LVL BYPASS switch does NOT remove the upper and lower Letdown flow limitations. This function is controlled by the LTDN LIMIT BYPASS switch, which is used to restore Charging and Letdown.
D is incorrect. The PZR LVL BYPASS switch will NOT prevent a low Pressurizer level from tripping the heaters.

References

LP PLC-01-C, R-4, Pg 27, PZR LVL Bypass function
OP-2207, St 4.2.5b, align switch for plant cooldown

Comments and Question Modification History

In Answer 'C' replaced '65% Pressurizer level' with '40% Pressurizer level' Pressurizer level is maintained at 40% after a plant shutdown. 65% is the normal 100% power level.

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

Generic **KIA** Selected

NRC KIA Generic System 2.1 Conduct of Operations

Number 2.1.27 RO 2.8 SRO 2.9 CFR Link (CFR: 41.7)
Knowledge of system purpose and or function.

Question #: 87 | **Question ID: 8000035** | ☐ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

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

A Large Break LOCA occurred approximately 20 hours ago, resulting in confirmed, significant core damage. Numerous Process and Area Radiation Monitors in the Aux Building and Enclosure Building are in alarm and indicating >10 Rem per hour dose rates in various locations.
Since that point, all equipment has been operating as expected. All procedural steps of EOP 2532, Loss of Coolant Accident, have been successfully implemented, up to and including opening all the ESF Room Sump Pump breakers.

Approximately 4 hours later, the "SI RM 'A' SUMP LEVEL HI" annunciator alarms (first time for this alarm).

How will this affect continued plant operations and what action must be directed?

☑ **A** Rising water level in the " A ESF Room will eventually flood out the " A HPSI Pump.
Direct a PEO to close the breakers for both sump pumps associated with the " A ESF Room for 10 minutes, then reopen the breakers, in accordance with EOP 2541, Appendix 44, Monitoring SI Room Leakage.

☐ **B** The additional Loss of Coolant will result in a loss of RCS inventory for Sump Recirculation.
Direct a PEO to close the breakers for both sump pumps in the "A" ESF Room, inspect " A SI Room, and attempt to isolate the leak, in accordance with EOP 2532, Loss of Coolant Accident.

☐ **C** Rising water level in the " A ESF Room will eventually flood out the " A HPSI Pump.
Dispatch a PEO to the " A ESF Room. If water level is high enough to cause damage to any SI Pump, deenergize the entire train of Safety Injection, in accordance with the Annunciator Response Procedure.

☐ **D** The additional Loss of Coolant will result in a loss of RCS inventory for Sump Recirculation.
Immediately stop all SI Pumps, close the affected Containment Sump Outlet Isolation Valve, and attempt to determine and isolate the source of the leak, in accordance with EOP 2532, Loss of Coolant Accident.

Justification

A is correct. If any SI Room Sump High Level annunciator alarms after the associated breakers are open, then EOP 2532 directs the crew to Refer To Appendix 44, which requires both sump pump breakers for a given room to be closed for 10 minutes, then reopened.
B is incorrect. Appendix 44 requires the crew to attempt to enter the room and identify the source of the leakage ONLY after the sump high level annunciator alarms a second time and the SI Pumps are stopped. In this case, it would be inappropriate to place a PEO in a high radiation area unnecessarily.
C is incorrect. Although it may be appropriate, depending on circumstances, the Annunciator Response Procedure (ARP 2590E-109) does NOT address deenergizing all SI Pumps for a high water level. It merely provides guidance for pumping the sump.
D is incorrect. Appendix 44 (NOT EOP 2532) provides guidance for stopping SI Pumps and closing the associated Containment Sump Outlet valve; however, this is performed ONLY after the second sump high level annunciator.

References

EOP-2532, SI Sump guidance, AOP-2590E, "A" SI Sump Alarm guidance.
EOP-2541, App. 44, SI Sump pumping guidance.

Comments and Question Modification History

In the stem, replaced 'several rem per hour' with '>10 Rem per hour'. To ensure examinees understood the magnitude of the high radiation areas in the Auxiliary Building.

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

Number A2.03 RO 3.3 SRO 3.7 CFR Link (CFR: 41.5 145.5)

Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: System leakage

Question #: **88** | **Question ID:** **53484** | ☐ **RO** | ☑ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

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

The plant has just entered Mode 4 on Shutdown Cooling with Reactor Coolant System (RCS) temperature at 220 °F and pressure at 245 psia.

- The 'A' and 'B' RCPs are operating.
- Both PORV LT/OP Setpoint Selector Switches are in "LOW".
- All four SIT outlet isolation valves have been closed and their key-lock Override Selector Switches are in "OVERRIDE".
- All other plant systems and components are aligned per the existing plant conditions.

Then, Pressurizer pressure transmitter P-103-1 fails HIGH.

Which one of the following describes the consequences of this failure, and an action that must be directed, to mitigate the pressure transmitter failure?

☐ **A** Two (2) SIT outlet valves will open. Direct the SIT outlet valves be placed in "OPERATE" and closed.

☐ **B** 2-SI-651 will close, isolating Shutdown Cooling. Direct 2-SI-651 be overridden open and restore SDC flow.

☐ **C** PORV 2-RC-402 will open. Direct the Setpoint Selector Switch for 2-RC-402 be put in "HIGH".

☐ **D** Auto Aux. feedwater will actuate in 205 seconds. Direct AFAS be overridden on Facility One

Justification

C - CORRECT; When P-103-1 fails high, it will cause PORV 2-RC-402 to open due to the setpoint control switch being in the "LOW" setting. In order to close the valve (or the block valve to just isolate the open PORV) the setpoint selector switch must be placed in "HIGH" position.
A -WRONG; With the SIT outlet valve control switches in the "override" position, the failed pressure transmitter will have NO effect on the SIT outlet isolation valves.
B -WRONG; There is no longer an auto close feature on high pressure for 2-SI-651 or 2-SI-652. It was removed in a previous plant design change to prevent an inadvertent loss of SDC due to a failed instrument.
D - WRONG; This is NOT the correct pressure switch for an AFAS. The failed instrument is a "control channel", where as the AFAS requires a signal from the "safety channels".

References

ARP-2590B-209, LT/OP HI/LO alarm response.
PORV Logic showing input from PT-103/103-1 and LP RCS-01-C; Description of PT-103.

NO Comments or Question Modification History at this time.

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

Number A2.03 **RO** 4.1 **SRO** 4.2 **CFR Link** (CFR: 41.5 143.5 145.3 145.13)

Ability to **(a)** predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: PORV failures

Question #: **89** | ***Question ID:*** *8000037* ☐ ***RO*** ☐ ***SRO*** ☐ **Student Handout?** ☐ **Lower Order?**

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

The crew has just entered MODE 4 and is making preparations to place SDC in service. Suddenly, numerous annunciators alarm, several Facility 1 components change position, and several Facility 1 indicators are deenergized.

Within 5 minutes after the initial event, the following report is provided to you:

The " A RBCCW Heat Exchanger indicates 9,000 gpm Service Water flow.
Letdown indicates '0' flow.
" A ESF Room Cooler indicates 60 gpm flow.
"A" SDC Heat Exchanger indicates 2,000 gpm RBCCW flow.
The SFP Cooling Low Flow annunciator is in alarm.
#1 S/G level indicates 65% and is slowly rising.
#2 S/G level indicates 65% and is stable.
Indicating lights for all Bus 24C breakers are out.

Which of the following is the appropriate procedure for you to enter for this event?

A Loss of 120 Volt AC Vital Instrument Panel, VA-10, AOP 2504C

☐ **B** Restoring DC and Vital Instrument AC Buses, EOP 2541, Appendix 29

☐ **C** Loss of Vital 125 VDC Instrument Panel, DV-10, AOP 2506A

☐ **D** Loss of All AC Power During Shutdown Conditions, AOP 2583

Justification

C is correct. The above indications are associated with a loss of either Vital Bus 201A of Vital DC Panel, DV-10. The appropriate AOPs may be used while in lower modes; EOPs are NOT used in lower modes.
A is Incorrect. The loss of Vital AC Instrument Panel, VA-10, will result in the loss of numerous indications and controls and may appear similar to the loss of DV-10. However, the indicating lights for 24C are NOT affected by this power loss.
B is incorrect Appendix 29 only provides actions for energizing Bus 201A from Battery 201A. Additionally, it would be inappropriate to enter Appendix 29 directly while in MODE 4.
D is incorrect. Although a loss of control power is indicated, it is NOT AC control power (VR-11/21). A loss of DC control power would cause these indications.

References

AOP-2506A, DV-10 Load List (24C Control Power)

Comments and Question Modification History

In the stem, changed 'MODE 5' to 'MODE 4'. Shutdown Cooling preps will be made in MODE 4, not MODE 5.

NRC K/A System/E/A **System** 063 DC Electrical Distribution System

Generic KIA Selected

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

Number 2.4.4 **RO** 4.5 **SRO** 4.7 **CFR Link** (CFR: 41.10 143.2 145.6)

Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Question #: **90** **Question ID:** **62764** ☐ **RO** ☑ **SRO** ☑ Student Handout? ☐ Lower Order?

Rev. **7** ☑ Selected for Exam **Origin:** **Bank** ☐ Past NRC Exam?

While operating at 100% power, normal conditions, the BOP was directed to pump the Containment Sump to complete filling an Aerated Waste Drain Tank for processing. After pumping the Containment sump, BOTH Containment Sump Inboard and Outboard Isolation Valves, 2-SSP-16.1 and 2-SSP-16.2, have dual indication.
A PEO sent to investigate the Containment Sump Outboard Isolation Valve, 2-SSP 16.2, has determined it is partially open. The valve will NOT close and must be disassembled for repair.

Simultaneously, a Containment entry is made to visually inspect the Containment Sump Inboard Isolation Valve, 2-SSP-16.1.
The valve will NOT fully close and also must be disassembled for repair.
The inner Containment door has been open 30 minutes while personnel are in Containment.

Which of the following actions must be directed?

☐ **A** 1. Restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the next 30 hours.
2. Immediately initiate action to evaluate overall containment leakage rate and verify an air lock door is closed within 30 minutes. Restore the air lock to OPERABLE status within 24 hours.

☐ **B** 1. Ensure the other two required leakage detection systems are OPERABLE and obtain and analyze grab samples at least once per 24 hours; otherwise, be in COLD SHUTDOWN within the next 36 hours.
2. Restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the next 30 hours.

☐ **C** 1. Immediately initiate action to evaluate overall containment leakage rate and verify an air lock door is closed within 30 minutes. Restore the air lock to OPERABLE status within 24 hours.
2. Restore the inoperable valve to OPERABLE status, or isolate the affected penetration within 4 hours, or be in COLD SHUTDOWN within the next 36 hours.

☑ **D** 1. Restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the next 30 hours.
2. Restore the inoperable valve to OPERABLE status, or isolate the affected penetration within 4 hours, or be in COLD SHUTDOWN within the next 36 hours.

Justification

D is correct, CTMT Isolation Valves 2-SSP-16.1 and 16.2 are NOT OPERABLE in that they do NOT go full closed (LCO 3.6.3.1). At least one of the valves must be OPERABLE to satisfy the definition of CONTAINMENT INTEGRITY (3.6 1.1) (Note: examinees must recognize that failure to meet 3.6.3.1 is the basis for considering the surveillance criteria of 3.6.1.1 NOT met.)
A is incorrect The Containment Air Lock is OPERABLE. (LCO 3.6.1.3)
B is incorrect The containment sump level monitoring system is OPERABLE (LCO 3.4.6.1)
C is incorrect. The Containment Air Lock is OPERABLE. (LCO 3.6.1.3)

References **Provided**

TSAS 3.6.1.1 (CTMT Integrity) and 3.6.3.1 (CTMT Isolation Valves)
Selected Unit 2 Technical Specifications without Bases.

NO Comments or Question Modification History at this **time.**

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

Generic KIA **Selected**

NRC KIA Generic System 2.2 Equipment Control

Number 2.2.22 RO 4.0 SRO 4.7 CFR Link (CFR: 41.5 / 43.2 145.2)

Knowledge of limiting conditions for operations and safety limits.

Question #: **91** | **Question ID: 8000038** | ☐ **RO** | ☑ **SRO** | ☐ Student Handout? | ☑ Lower Order?

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

While operating at 100% power, the plant suddenly trips due to grid instabilities, concurrent with a loss of offsite power. During the performance of EOP 2525, you direct the Balance of Plant operator to reduce both ADV setpoints to the lower end of the Steam Generator pressure control band.

Which one of the following is the basis for directing this action?

☐ **A** To minimize the release of activity due to a subsequent Steam Generator Tube Rupture.

☐ **B** To ensure Steam Generators are available for a subsequent Loss of Coolant Accident.

☐ **C** To promote the development of natural circulation for a subsequent Station Blackout.

☑ **D** To assist in determining whether or not a Main Steam Safety is actually stuck open.

Justification

D is correct. Due to the low blowdown setpoint (~880 psia) for the Main Steam Safeties, it's possible for safeties to be open post-trip when the Condenser Steam Dumps are not available and the ADV setpoints are at 920 psia. Lowering the ADV setpoint to the lower end of the control band (880 psia) will assist in determining whether a safety is stuck open. OP 2260, Attachment 1, step 5b.
A is Incorrect. Although it may lower the release rate for some period of time, lowering SIG pressure during a SGTR will NOT reduce the release of activity from a SGTR. Ultimately, the same amount of activity will be released.
B is incorrect. Lowering the setpoint for the ADVs does not provide any additional assurance that S/Gs will be available during a LOCA.
C is incorrect. Although natural circulation may develop slightly sooner due to the larger delta T from lowering SIG pressure and Tc, the difference is insignificant.

References

OP 2260, Attachment 1, step 5b.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A System 041 Steam Dump System (SDS) and Turbine Bypass Control

Generic WA Selected

NRC KIA Generic System 2.2 Emergency Procedures /Plan

Number 2.4.18 RO 3.3 SRO 4.0 CFR Link (CFR: 41.10 143.1 145.13)

Knowledge of the specific bases for EOPs.

Question #: 92 | **Question ID:** 8000039 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☑ Lower Order?

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

The plant is operating at 100% power with 6 Condensate Polishing Demineralizers in service. Three of them are in Amine Form. Suddenly, a large Condenser Tube Leak develops in the " A Waterbox.

After the crew isolates the " A Waterbox, which of the following additional actions must be directed and why?

☐ **A** Raise Blowdown flow as requested by Chemistry and, per AOP 2616, Condenser Tube Leak, throttle CPF Demineralizer Bypass, 2-CNM-2, to isolate the demineralizers in the Amine Form.
Demineralizer resin damage may result from a chemical reaction between the Amine and seawater.

☑ **B** Secure Blowdown flow and, per AOP 2575, Rapid Downpower, reduce power to a level that will allow removing the three Amine Form demineralizers from service.
The Amine Form demineralizers will release more Ethanolamine when contaminated with seawater.

☐ **C** Secure Blowdown flow and, per AOP 2575, Rapid Downpower, reduce power to a level that will allow removing the three Mixed Bed demineralizers from service.
The Amine Form demineralizers are more efficient at removing contamination due to seawater.

☐ **D** Raise Blowdown flow to the maximum flow allowed and place the remaining Mixed Bed demineralizer in service to maximize the efficiency of the Condensate Polishing Facility.
The demineralizers, in any form, are very efficient at removing contamination due to seawater.

Justification

B is correct. Seawater interacting with a demineralizer in the Amine Form will cause Ethanolamine (ETA) to be released from the demineralizer more rapidly than normal. If ETA levels are increased significantly enough, then the levels at the Blowdown discharge will exceed NPDES limits. In order to prevent exceed any environmental limits, blowdown is secured and power level is reduced to allow removing the demineralizers in the Amine Form from service.
A is incorrect. If there were NO demineralizers in the Amine Form, then Blowdown flow would be raised to help eliminate contaminants in the S/Gs more quickly. Blowdown flow WILL be raised after the Amine Form demineralizers are removed from service. Additionally, CPF will NOT be bypassed with seawater contamination of the Condensate System.
C is incorrect. Blowdown flow will be secured and power will be reduced, but NOT for removal of the Mixed Bed demineralizers.
D is incorrect. If there were NO demineralizers in the Amine Form, then Blowdown flow would be raised to help eliminate contaminants in the S/Gs more quickly. Additionally, placing one additional mixed bed demineralizer in service will not necessarily reduce contaminations levels any quicker; it will only slightly lower the ultimate contamination levels of the other in-service demineralizers.

References

AOP-2516, Discussion on Cond. Tube Leak impact on Amine bed demins.

NO Comments or Question Modification History at this time.

NRC K/A System/E/A System 056 Condensate System

Number A2.05 RO 2.1 SRO 2.5* CFR Link (CFR: 41.5/43.5/45.3/45.13)

Ability to (a) predict the impacts of the following malfunctions or operations on the Condensate System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Condenser tube leakage

Question #: **93** | **Question ID:** **8000040** | ☐ **RO** | ☑ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The plant has tripped from 50%. EOP 2525 has been completed.

The following conditions exist:
- Reactor power is 3 X 10 E-5% and lowering with all but one CEA inserted
- 24D is deenergized due to a bus fault
- All other buses are energized by the RSST
- Pressurizer level is 18% and lowering slowly
- Pressurizer Pressure is 1870 psia and lowering slowly
- RCS temperature is 530°F and stable
- SG levels are 37% and slowly rising
- SG pressures are 900 psia and stable (supplied by " A MFP)
- CTMT pressure is 0.1 psia and stable
- SIAS, CIAS, and EBFAS have NOT been actuated.
- " A HPSI Pump is tagged out for PMs.
- CTMT Sump level is 48% and stable
- RBCCW Radiation Monitor, RM 6038, is in alarm
- RBCCW Surge Tank indicates 100% on C-06
- Aux Building PEO reports that the in-service Aerated Waste Drain Tank and the Aux Building -45'6" sump are both overflowing.
- HP reports that a lot of water is coming from the overhead of the East 38'6" level.
- All other equipment is operating as expected

Which one of the following safety functions are NOT satisfied for the EOP being entered?

☐ **A** Reactivity Control

☐ **B** RCS Inventory Control

☐ **C** RCS and Core Heat Removal

☑ **D** Containment Isolation

Justification

D is correct. The indications presented indicate an Intersystem LOCA into the RBCCW System which is overflowina into the Aerated Waste System. As a result of the RBCCW Radiation Monitor alarm, the Containment Isolation Safety Function is NOT met.
A is incorrect. As long as Reactor power is less than 10 E-4%, then, even with one CEA stuck out, the Reactivity Control Safety Function is met.
B is incorrect. Although Pressurizer level is less than 20% and is lowering, RCS Inventory Safety Function is met as long as Safety Injection meets the Stop/Throttle criteria of EOP 2532. Even without the "A" or "C" HPSI Pumps, the "B" HPSI Pump is available for injection if required.
C is incorrect. Although SG levels are less than 40%, they are being restored; therefore, RCS Heat Removal is met. With RCS temperature at 530°F and stable, Core Heat Removal is also met.

References

EOP-2532, Reactivity Safety Function and St. 7, RBCCW leakage indication.

NO Comments or Question Modification Historv at this time.

NRC K/A System/E/A System 068 Liquid Radwaste System (LRS)

Generic WA Selected

NRC K/A Generic System 2.4 Emergency Procedures /Plan

Number 2.4.21 RO 4.0 SRO 4.6 CFR Link (CFR: 41.7 / 43.5 145.12)

Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Question #: **94** | ***Question ID:*** ***1000176*** | ☐ ***RO*** | ☑ ***SRO*** | ☐ Student Handout? | ☐ Lower Order?

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

The crew has entered EOP 2540C1, Functional Recovery of RCS Inventory Control, Success Path IC-2: Safety Injection, as a result of a small break LOCA concurrent with an Excess Steam Demand event in containment on #1 S/G. All equipment is operating as expected and all required actions to this point have been performed.

The following conditions exist:

- Reactor power is 2 E-6% and lowering
- No RCPs are operating
- Pressurizer pressure is 825 psia and lowering
- Pressurizer level is 0%
- RVLMS indicates 80% on both channels and stable
- CET temperature is 468°F and lowering
- #1 S/G level is 160 inches (wide range) and lowering with feedwater secured
- #2 S/G level is 48% and rising with both motor driven auxiliary feedwater pumps running
- #1 S/G pressure is 230 psia and lowering
- #2 SIG pressure is 500 psia and lowering
- Containment pressure is 17 psig and lowering

Which of the following actions must the US direct to ensure proper operation of the Safety Injection System?

☐ **A** Throttle HPSI to a minimum of 40 gpm using the HPSI Injection Valves.

☐ **B** Place all HPSI Pump control switches to 'Start' then to 'Pull-To-Lock'.

☐ **C** Ensure HPSI flow is being maintained within the SI Flow Curve.

☐ **D** Maximize safety injection by starting the third HPSI Pump.

Justification

"C" is correct. The given conditions do NOT meet SI throttle stop criteria due to a pressurizer level of less than 20%; therefore, EOP 2540C1 requires the operator to ensure adequate safety injection flow per the SI flow curve.
"A" is incorrect. Pressurizer level is less than 20%. This does NOT meet HPSI throttle criteria.
" B is incorrect. The HPSI stop criteria has the same requirements as the HPSI throttle criteria, but adds requirements for reactor power.
"D" is incorrect. The procedure does not allow starting a third HPSI pump to increase SI flow.

References

EOP-2540C1, Section IC-2, HPSI Throttle and/or Stop Criteria

NO Comments or Question Modification History at this time.

NRC K/A System/E/A System 2.1 Conduct of Operations

Generic KIA Selected

NRC KIA Generic System 2.1 Conduct of Operations

Number 2.1.9 RO 2.9 SRO 4.5 CFR Link (CFR: 41.10 / 45.5 145.12 145.13)

Ability to direct personnel activities inside the control room.

Question #: **95** | *Question ID:* *8000048* ☐ *RO* ☐ *SRO* ☐ Student Handout? ☑ Lower Order?

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

Approximately 45 minutes ago an Alert was declared in Unit 2 due to a fire in the Spent Fuel Pool railway access area.
The SERO is fully manned but the CR-DSEO has not yet been relieved by EOF.
The Waterford Fire department is on site and has taken control of all fire fighting activities, but has requested additional fire stations be called in to assist.

Which of the following personnel must the CR-DSEO notify to ensure site access for the additional Waterford Fire Department personnel when they arrive?

☐ **A** Waterford Fire Chief.

☐ **B** Waterford Police Dispatch.

☐ **C** Manager of Resources.

☑ **D** Manager of Security.

Justification

D - Correct; Manager of Security (referred to as the Security Shift Supervisor outside of SERO activation) is responsible for site access, even during a SERO event where outside agencies are involved.
A - Wrong; The Fire Chief is responsible for the arriving personnel once they are ON site.
B - Wrong; They are notified and are responsible for access TO the site, but NOT "site" access.
C -Wrong; The MOR is responsible for getting additional company personnel to the site, and working with security to get them into the site.

References

AOP-2559, Notify for additional personnel request.

~ N Comments or Question Modification History at this time.

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

Generic WA Selected

NRC KIA Generic **System** 2.1 Conduct of Operations

Number 2.1.13 **RO** 2.5 **SRO** 3.2 **CFR Link** (CFR: 41.10 / 43.5 145.9 145.10)

Knowledge of facility requirements for controlling vital I controlled access.

Question #: **96** | ***Question ID:* 8000041** | ☐ **RO** | ☑ **SRO** | ☐ Student Handout? | ▲ Lower Order?

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

The plant is at 100% power, steady state, with all components and equipment functioning as designed.

The following personnel status presently exists:
* The Shift Manager (SM) is in the SM office discussing present plant conditions with the Operations Manager.
* The Unit Supervisor (US) is at his desk in the Control Room.
* The Reactor Operator (RO) has gone down the hall from the Control Room to use the rest room.
* The Balance-Of-Plant Operator (BOP) is at the controls, adjusting Main Generator VAR loading.
* A Qualified PEO standing an RO training watch for License Operator Initial Training (LOIT) is observing the BOP.
* An SRO Licensed Instructor, standing an SM training watch to activate his license for fuel handling operations, is at the STA desk.
* A spare RO qualified operator is in the Dosemetry Office exchanging his TLD.

Then, the plant trips due to state-wide blackout and the following conditions now exist:
- Four **(4)** CEAs indicate fully withdrawn on the CEA Mimic.
- The " A EDG is running but the output breaker did not close.
- Facility One Vital and Non-vital AC busses are dead.

All other plant components and systems are operating as designed for the existing conditions.

Which of the following are the appropriate actions for the US to take under the existing conditions?

☑ **A** Instruct the BOP to perform the BOP Immediate Actions of EOP 2525, Standard Post Trip Actions. The SM takes the "oversight" role and the US performs the RO Immediate Actions of EOP 2525, Standard Post Trip Actions.

☐ **B** Instruct the BOP to perform the BOP Immediate Actions of EOP 2525, Standard Post Trip Actions. The SM performs the RO Immediate Actions of EOP 2525, Standard Post Trip Actions, under the US supervision.

☐ **C** Instruct the BOP to perform the BOP Immediate Actions of EOP 2525, Standard Post Trip Actions. Instruct the SRO Licensed Instructor to supervise the BOP Trainee performing the RO Immediate Actions of EOP 2525, Standard Post Trip Actions.

☐ **D** Instruct the spare qualified RO to come to the Control Room and perform the BOP Immediate Actions of EOP 2525, Standard Post Trip Actions.
Instruct the BOP to perform the RO Immediate Actions of EOP 2525, Standard Post Trip Actions.

Justification

A - CORRECT; the Immediate Actions of EOP-2525 must be completed as quickly and efficiently as possible by personnel trained and qualified to perform them. For EOP-2525, the SM will take the oversight role of the US so the US may perform the duties of an RO/BOP.
B - WRONG; the SM should never drop to the role of "Control Board Operator" if the US and an additional qualified RO/BOP are available. The SM must remain in an oversight position to ensure those duties are performed.
C - WRONG; the Licensed Instructor, although very qualified to supervise the training and evaluation of a licensed operator, is NOT qualified to perform those duties in the US role, unless qualified as a US. This qualification requires an "active" SRO license, which this Instructor must NOT have if the individual is standing a "training" watch to get qualified for Fuel Movement.
D - WRONG; the US can NOT delay the performance of EOP-2525 until a qualified RO can be summoned from Work Control.

References

0P-2260, Command and Control requirements for US and SM

NO Comments or Question Modification History at this time.

NRC KIA System/E/A System 2.2 Equipment Control

Generic KIA Selected

NRC KIA Generic System 2.2 Equipment Control

Number 2.2.2 RO 4.6 SRO 4.1 CFR Link (CFR: 41.6 141.7 145.2)

Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels

Question #: **97** **Question ID: 8000052** ☐ **RO** ☑ **SRO** ☐ Student Handout? ☑ Lower **Order?**

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

Which of the following evolutions is considered an Infrequently Conducted or Complex Evolution (ICCE)?

☐ **A** The quarterly surveillance for performing CEA partial movement

☐ **B** A reactor startup shortly after an uncomplicated reactor trip

☐ **C** The plant coastdown prior to the plant shut down for refueling

☐ **D** Local Leak Rate testing of all Containment penetrations

Justification

B is correct. OP 2202, Reactor Startup, is labeled as an ICCE regardless of when it is being performed. This procedure meets two of the four following requirements to categorize it as an ICCE:
- Not specifically addressed by an existing normal or abnormal procedure (NOT met)
- Involve complex sequencing or unusual plant configurations (Met)
- Are performed at a frequency that may undermine personnel proficiency (Met)
- Require the use of special test procedures in conjunction with existing procedures. (Not met)

A is incorrect. Although this surveillance is performed only every three months, it is considered NOT to meet any of the four criteria.
C is incorrect. Although a plant coastdown is performed, at most, every 18 months, other evolutions similar in nature (e.g. plant downpower) are performed frequently enough to ensure personnel proficiency is maintained. Additionally, this is NOT considered an unusual plant configuration and a normal operating procedure exists; therefore, this does NOT meet any of the 4 criteria.
D is incorrect. Although local leak rate testing is performed every 18 months, there are similar evolutions that ensure personnel proficiency is maintained. Even though plant configuration is changed to perform each test, the changes are NOT complex. Additionally, the surveillance procedure is very specific on how to conduct each test; therefore, this is NOT considered and ICCE.

References

OP-AA-106, infrequently conducted or Complex Tests or Evolutions (ICCE), Pg 9, St 5.3.1b

Comments and Question Modification History

Replaced original question. Met the K/A, but had difficulty with the stem. Very difficult to understand the stem versus low level of difficulty when fixed. 11114108

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

Generic K/A Selected

NRC KIA Generic System 2.2 Equipment Control

Number 2.2.7 **RO** 2.9 **SRO** 3.6 CFR Link (CFR: 41.10 143.3 145.13)

Knowledge of the process for conducting special or infrequent tests.

Question #: **98** | **Question ID: 8000043** | ☐ **RO** | ☑ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

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

A Fuel Handling Accident has occurred inside Containment as evidenced by several Containment process and area radiation monitors rising or in alarm. All personnel were immediately evacuated.

In order to limit personnel exposure, the Shift Manager must ensure which of the following are performed?

☑ **A** The isolation of Containment must be completed no more than 30 minutes after the event.

B Containment Purge valves, AC-4, 5, 6, and 7 must be closed within 50 minutes of the event.

☐ **C** The Transfer Tube Isolation Valve, RW-280, must be closed within 30 minutes after the Transfer Carriage is in the SFP.

☐ **D** At least one train of Control Room Air Conditioning is operating in the Recirculation mode within 5 minutes of the event.

Justification

A is correct. Per AOP-2577 (Fuel Handling Accident), Rev. 008-02, Section 1.0, Containment must be isolated within 30 min. of the event.
B is incorrect. The Containment Purge Valves would be closed for a Fuel Handling Accident in Containment, but they would likely have closed automatically. However, if the auto actuation has not occurred, they must be manually closed immediately.
C is incorrect. The Transfer Tube Isolation Valve, RW-280, is not included in the components that must be closed for Containment Isolation in a Fuel Handling Accident as the opening is low in the refuel pool. This valve would be closed for a loss of Refuel Pool level and takes about 30 minutes to close.
D is incorrect. The calculation for the Control Room radiological exposure following a Fuel Handling Accident is based on having at least one Control Room Air Conditioning train operating in the Recirculation mode within 60 minutes of the event. The calculations used to require recirc mode within 10 minutes. However, new calculations recently changed the requirement to 50 minutes with a 10 minute penalty for other system alignment complications. (AOP-2577 had NOT yet been updated when the question was written.)

References

AOP-2577 (Fuel Handling Accident), Rev. 008-02, Section 1.0

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.3 Radiation Control

Generic KIA Selected

NRC KIA Generic **System** 2.3 Radiation Control

Number 2.3.13 **RO** 3.4 **SRO** 3.8 **CFR Link** (CFR: 41.12 143.4 145.9145.10)

Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Question #: **99** **Question ID:** 80578 ☐ RO ☑ SRO ☐ Student Handout? ☐ Lower Order?

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

The plant has tripped from 100% power due to a Large Break Loss of Coolant Accident.

Which of the following describes a concern if the Containment Spray System is secured less than 4 hours after the event?

☐ **A** Containment equipment would be adversely affected due to operating in a much harsher environment.

☑ **B** Early termination of Containment Spray would result in higher Iodine concentrations in Containment.

☐ **C** When initiated, Sump Recirculation will NOT provide adequate long term heat removal from the RCS.

☐ **D** The resultant increase of Hydrogen in Containment could potentially jeopardize Containment Integrity.

Justification

B - Correct; EOP 2532 Tech Guide requires Containment Spray to operate for at least 4 hours for Iodine scrubbing. Tech Spec bases requires Containment Spray to be operable for Iodine scrubbing.
A - Wrong; The CTMT environment is much worse during an ESD event, but the requirement for securing CTMT Spray flow operation during that accident is based only on CTMT pressure.
C - WRONG, Securing CTMT Spray flow does NOT secure Sump Recirc flow from any SI pumps through the CTMT Spray coolers.
D - WRONG; Hydrogen production at this time in the event would be negligible, unless core uncovery occurred. In that case, Tech. Support guidance be required as the event would have progressed beyond the design criteria for which the EOPs are based.

,References

EOP-2532, St. 60, CTMT Spray Stop Criteria
Tech Spec Bases for CTMT Spray

N O 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** 2.2 Emergency Procedures /Plan

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

Knowledge of the specific bases for EOPs.

Question #: **100** | ***Question ID:*** **8000042** | ☐ ***RO*** | ☑ ***SRO*** | ☐ Student Handout? | ☑ Lower Order?

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

A serious event has occurred at Millstone Station and the Station Emergency Response Organization (SERO) is being fully manned.
Access to the station has been restricted to ALL offsite personnel, including SERO.

Which of the following describes where offsite SERO personnel, who normally report to EOF, must now report?

☑ **A** The National Guard Base (or Camp) in East Lyme.

☐ **B** The Waterford Emergency Dispatch Center

☐ **C** The Waterford Town Hall assembly room.

☐ **D** The East Lyme/Niantic Police Building.

Justification

A - CORRECT; When access to the site is unavailable, SERO personnel are instructed to report to the "Remote Mustering Area", which is the local National Guard Base, presently named "Camp Rell" (the base is always named after the present Connecticut governor).
B -WRONG; This is a logical choice as it is where the Waterford town personnel would be controlled from during the emergency. It also has a direct communication with both Control Rooms and all key state agencies.
C - WRONG; This is where the Waterford Town personnel would report and be dispatched from during an emergency. It has preexisting communication links to key town areas including Waterford Emergency Dispatch.
D - WRONG; This is the old location of the Millstone Energy Center, where people could report, log in to the Millstone LAN and get real time plant data. It was recently sold to the town of East Lyme and is now their Police/Dispatch Station. The building also has a direct communications link with each unit Control Room and all key state agencies. However, ALL Millstone controlled equipment has been completely removed.

References

Q-57, OP-2302A, CEDS Interlocks

NO Comments or Question Modification History at this time.

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

Generic **KIA** Selected

NRC KIA Generic System 2.4 Emergency Procedures /Plan

Number 2.4.28 RO 3.2 SRO 4.1 **CFR** Link (CFR: 41.10 143.5 145.13)
Knowledge of procedures relating to a security event (non-safeguards information),