Question #: **1** **Question ID: 1653292** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☐ **Lower Order?**
Rev. 8 ☑ **Selected for Exam** **Origin: Bank** ☐ **Past NRC Exam?**

The plant is operating normally at 100% power when the following occurs:
- CONVEX orders an Emergency Generation Reduction 600 MWe
- The required procedural steps are successfully carried out
- The plant is stabilized at approximately 600 MWe output.

Then, the plant trips due to a trip of the 'A' RCP.

Which one of the following plant conditions will the BOP observe, that will require an action to be taken during trip recovery?

☐ **A** There is a much higher probability that one or more Main Steam Safety Valves will open on the plant trip.

☑ **B** All steam dump valves to the condenser will remain partially open regardless of how low Tavg drops.

☐ **C** The Turbine Bypass Valve ("A" Condenser Dump) will fully close at a lower Main Steam header pressure.

☐ **D** Only the two ADVs and the Turbine Bypass Valve ("A" Condenser Dump) will fully open on the trip.

Question Misc. Info: MP2*LOIT/LORT* RRS, SDBV, 2557, NRC-2016

Justification

A - WRONG; the "quick open" signal is still sent to all 6 dump valves on the trip. There is no change in how much steam they must dump on the trip to prevent MSSVs from opening, and the valves already have a head start by being partially open.
Plausible; Examinee may believe that because the Condenser Dumps Valves are in manual control, per the AOP, they will be delayed in responding post-trip, which is partially true. However, even though the AOP stresses the need to shift TIC-4165 to auto immediately following a plant trip to allow the steam dump valves to lower RCS Tavg as designed, this is not done to prevent the opening of MSSVs.

B - CORRECT; AOP-2557 places TIC-4165 in manual to open the Condenser Dumps Valves. Therefore, as RCS Tavg lowers on the trip, the signal will have NO effect on their position.

C - WRONG; The Turbine Bypass valve will be held open by the larger demand signal from TIC-4165 (in Manual).
Plausible; Examinee may believe that because the AOP keeps the 'A' steam dump in automatic control, it will respond to the lowering Main Steam pressure post-trip.

D - WRONG; The trip signal will generate a "quick open" to all 6 steam dump valves, even with controller TIC-4165 in Manual.
Plausible; Examinee may believe that only the valves that are controlled automatically by main steam pressure will fully respond on the trip. Once the Q.O. Signal is reset (about 3 - 5 seconds post-trip), the steam dumps will close to the value of the TIC-4165 manual output signal.

References

AOP 2557

NO Comments or Question Modification History at this time.

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

Generic K/A Selected 3.7 3.7 (CFR: 41.7 / 45.5 / 45.6)

NRC K/A Generic **System**

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

Question #: **2** **Question ID: 2016034** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☐ **Lower Order?**

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

The reactor was tripped from 100% power due to a Pressurizer Safety valve failed OPEN.

The crew has transitioned to EOP 2532, Loss of Coolant Accident.

Which of the following describes the bases for the HPSI Throttle/Stop criteria, as it would pertain to this accident?

☑ **A** RCS subcooling margin is required to verify adequate Inventory Control.

☐ **B** RCS subcooling margin is required to verify core heat transfer to the SGs.

☐ **C** Pressurizer level is required to verify adequate Inventory Control.

☐ **D** Pressurizer level is required to verify core heat transfer to the SGs.

Question Misc. Info: MP2*LOIT EOP, 2532, NRC-2016

Justification

"A" - CORRECT; Subcooling is a key parameter that is lost in a vapor space accident, but it is still required to ensure indications of PZR level are correctly interpreting RCS inventory.

"B" - WRONG; In a worse case vapor space accident, core heat transfer to the SGs will be accomplished through reflux cooling, which makes subcooling irrelevant.
Plausible; Examinee may recall that adequate subcooling is a requirement in the EOP to verify Natural Circulation, for the purpose of ensuring adequate heat transfer from the core to the SGs.

"C" - WRONG; Pressurizer level will be artificially high in this type of accident, giving false indication of actual RCS inventory.
Plausible; Examinee may recall that Pressurizer level **is** a requirement of the safety function for verification of Inventory Control.

"D" WRONG; Although pressurizer level is used to indicate in some events that there is enough water in the RCS to utilize the SGs as a heat sink, in this event it will be a false indication due to the bubble formation in the core.
Plausible; Examinee may recall that pressurizer level is a criteria for ECCS flow throttling, and that OP 2260 Attachment 4 EOP 2532, "Loss of Coolant Accident Implementation Guide", allows for throttling injection flow when approaching the upper control band for the PZR (but only in the case of Shutdown Margin).

References

TG2532 r29

NO Comments or Question Modification History at this time.

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

Number AK3.05 **RO** 4.0 **SRO** 4.5 **CFR Link** (CFR 41.5,41.10 / 45.6 / 45.13)

Knowledge of the reasons for the following responses as they apply to the Pressurizer Vapor Space Accident: ECCS termination or throttling criteria

Question #: 3 **Question ID:** 2016001 ☑ RO ☐ SRO ☐ Student Handout? ☐ Lower Order?
Rev. 0 ☑ Selected for Exam **Origin:** New ☐ Past NRC Exam?

The plant has been shutdown in preparation to start a refueling outage.

The crew has just begun an RCS cooldown to Mode 4 when numerous annunciators alarm and the following indications are observed:

- Pressurizer level = 35% and lowering at ~ 3%/minute.
- Pressurizer pressure = 1900 psia and lowering at ~ 25 psia/minute.
- Letdown flow = 28 gpm and stable.
- Charging flow = 132 gpm and stable.
- RCS temperature = 350°F and lowering at 30°F/hour.

Once the crew has stabilized RCS temperature, which one of the following procedures will provide the specific steps used to mitigate this event?

☑ **A** EOP 2532, Loss Of Coolant Accident.

☐ **B** AOP 2568, Reactor Coolant System Leak.

☐ **C** AOP 2568A, RCS Leak Mode 4, 5, 6 and Defueled.

☐ **D** OP 2207, Plant Cooldown, Att. 10, Conditional Actions.

Question Misc. Info: MP2*LOIT, 2532, EOP, NRC-2016

Justification

A - CORRECT; With the plant in Mode 3, LOCA indications require entry into EOP 2532, LOCA.
The following thumb rule applies:

- 1% PZR LVL = 1°F RCS Δ
- 1% PZR LVL = 67 gallons @ Normal Operating Temperature or 112 gallons Cold Shut Down

Therefore 30°F/Hr Cooldown@ 350°F = ≈ 0.5% PZR LVL / min = ≈ 45 gpm (using 90 gallons/% @ 350°F)
Therefore 2.5%/min is due to the RCS Leak = 90×2.5=180 gpm which is greater than the capacity of the charging system therefore LOCA vice RCS leak.

B - WRONG; Conditions indicate a leak greater than the capacity of charging, based on the existing cooldown rate (and thumb rules for effect on PZR).
Plausible; If the examinee used 1°F Δ RCS = 1 % PZR LVL = ≈ 0.5% PZR LVL / min => ≈ 2.5% / min due to RCS leak, using 1% PZR LVL ≈ 60 gallons @ Normal Operating Temperature => 150 gpm leak - 28 gpm Letdown = 122 gpm < Charging pump capacity.

C - WRONG; 2568A is not applicable yet, due to existing RCS conditions.
Plausible; The examinee may think that the new lower Mode RCS Leak procedure AOP 2568A may apply using the same leak rate calculation as in answer “B” above.

D - WRONG; 2207 Conditional Actions do apply, but they defer to 2532 for event mitigation based on existing RCS conditions (SIAS could not have been blocked).
Plausible; The examinee may assumes that OP 2207 the current procedure in use applies, due to the Attachment 10 “Plant Conditional Actions” which is only when the SDC cooling system is in service for all other conditions OP 2207 refers to the applicable AOP 2568 or 2568A.

References

EOP 2532

NO Comments or Question Modification History at this time.

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

Generic K/A 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 / 43.2 / 45.6)

Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Question #: **4** | **Question ID:** **1685734** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

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

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

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

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

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

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

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

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

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

Justification

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

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

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

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

References

RPS-01-C

NO Comments or Question Modification History at this time.

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

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

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

Question #: **5** | **Question ID:** 8000004 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 3 | ☑ Selected for Exam | **Origin:** Bank | ☐ Past NRC Exam?

A plant downpower was performed for Control Valve Testing.

Alarm C02/3 A-15 "CHARGING FLOW LO" is received and charging and letdown are secured.

The plant is being maintained constant at 90% power while corrective actions are being performed on the charging system.

Pressurizer level has peaked at 70%, but is now dropping at the steady rate of 5% per hour.

How long before pressurizer level lowers to the point where administrative requirements will require a plant trip?

☐ **A** 2 hour

☑ **B** 3 hours

☐ **C** 7 hours

☐ **D** 10 hours

Question Misc. Info: MP2*LOIT CVCS, AOP, 2512, NRC-2016

Justification

A - Wrong; This is the time it would take level to drop 10% [5%/hr x **2 hrs** = 10% drop], down to a level of 60%.
Plausible; Examinee may recall 10% drop criteria as meaning a total drop of 10% from existing level.

B - Correct; AOP-2512, Loss Of All Charging, requires the plant be tripped if pressurizer level drops to 10% below programmed Pressurizer level setpoint. At 90% power, programmed Pressurizer level has NOT started ramping down yet so setpoint is still 65%. Therefore, a trip is required when level drops to 55%. [5%/hr. X **3 hours =** 15% drop; 70% - 15% = 55% level; 65% setpoint - 55% level = 10% (maximum amount allowed below setpoint)].

C - Wrong; This assumes a trip is required at 35% [5%/hr x **7 hours** = 35% drop; 70% - 35% = 35% level], which is the minimum operating band in the procedures.
Plausible; When it was applicable, this would require action and eventual plant shutdown and/or trip.

D - Wrong; This assumes a trip is required when pressurizer level reaches 20% [5%/hr x **10 hrs** = 50% drop; 70% - 50% = 20% level] At this level, all heaters will automatically de-energize. A trip may be prudent, but the only administrative (Tech. Spec.) requirement is to shutdown upon loss of the Proportional Heaters.
Plausible; Examinee may recall that if PZR level drops to </= 20% on a dropped rod, a plant trip is immediately required due to loss of RCS pressure control.

References

AOP 2512

NO Comments or Question Modification History at this time.

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

Number AA1.02 **RO** 3.0 **SRO** 2.9 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the Loss of Reactor Coolant Makeup: CVCS charging low flow alarm, sensor, and indicator

Question #: 6 | **Question ID:** 2016039 | ☑ RO ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | **Origin:** New | ☐ Past NRC Exam?

The Reactor Operator has placed Shutdown Cooling in service with concurrent RCP Operations per OP 2207 “Plant Cooldown” Section 4.18, "Initiating SDC With Concurrent RCP Operation".

After the Operator established a maximum allowed cooldown using SDC, VA-10 was lost.

How does a loss of VA-10 affect the following interrelated systems for the given conditions?

☐ **A** The RBCCW pump Amps will be higher, Service Water pump Amps rise as the Heat Load from the RCS increases.

☐ **B** The RBCCW pump Amps will be lower, Service Water pump Amps lower as the Heat Load from the RCS decreases.

☑ **C** The RBCCW pump Amps remain unchanged, Service Water pump Amps lower as the Heat Load from the RCS decreases.

☐ **D** The RBCCW pump Amps remain unchanged, Service Water pump Amps rise as the Heat Load from the RCS increases.

Question Misc. Info: MP2*LOIT, 2564, RBCCW, 2330A, NRC-2016

Justification

A - WRONG; The procedure has the SDC HX RBCCW Outlet valves throttled, so even if the SSC valve fails open, it will not affect RBCCW flow.
Plausible; Examinee may focus on the SDC HX outlet valve failing open on loss of power, which would increase the RBCCW flow through the HX and raise pump amps.

B - WRONG; The loss of heat load on the RBCCW system due to the loss of letdown flow (VA-10 loss isolates letdown) will be negligible due to the much lower RCS temperatures and high flow demands through the SDC HX's.
Plausible; Examinee may focus on the loss of letdown flow and its normal effect on RBCCW heat loads and system flow demands.

C - CORRECT; A loss of VA-10 will cause 2-SI-306 "SDC HX Bypass" valve to fail OPEN and 2-SI-657 "SDC HX FLOW CNTL" valve to fail CLOSED in affect causing a loss of RHR, reducing the heat input to RBCCW System thus lowering the heat input to Service Water System. RBCCW pump Amps remain unchanged because RBCCW flows are manually throttled therefore RBCCW flows remain unchanged. The lowering heat input to the RBCCW system requires less SW system flow thereby lowering SW pump Amps.

D - WRONG; RCS heat loads on SDC will lower when SI-657 fails closed on loss of control power (VA-10 loss).
Plausible; Examinee may remember that SI-657 and SI-306 both fail either open or closed on a loss of control power, but not which direction they fail in. This answer would imply they fail in a "safe" way, to prevent a loss of heat sink on a power loss. However, the safety designation of the components is not based on RHR operation.

References

AOP 2572

NO Comments or Question Modification History at this time.

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

Number AK2.03 **RO** 2.7 **SRO** 2.7 **CFR Link** (CFR 41.7 / 45.7)

Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: Service water or closed cooling water pumps

Question #: **7** **Question ID: 1000103** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☐ **Lower Order?**

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

With the plant operating at 100% power, the following alarms are received:

- RBCCW HDR B PRESS LO.
- RBCCW HDR B FLOW HI.
- RBCCW SURGE TK LEVEL HI/LO.
- Various low flow annunciators for components supplied by "B" RBCCW header.

The cause of the indicated high flow on the "B" RBCCW header is a rupture ________________.

☐ **A** downstream of the RBCCW surge tank outlet orifice to the "B" RBCCW header.

☐ **B** on the RBCCW inlet piping to the "C" RBCCW heat exchanger.

☐ **C** between the "C" RBCCW pump discharge isolation and check valves.

☑ **D** on the RBCCW inlet piping to the letdown heat exchanger.

Question Misc. Info: LOIT, RBCCW, 2564, MB-05026, NRC2002, NRC-2016

Justification

A - WRONG; A rupture in the RBCCW supply header would NOT result in higher indicated header flow.
Plausible; Examinee may mistakenly believe that the surge tank outlet orifice is a flow device.

B - WRONG; The 'C' HX flow instrument is downstream of the heat exchangers.
Plausible; Examinee may incorrectly believe it is upstream.

C - WRONG; A rupture at this point would cause a low indicated flow due to water going out the break.
Plausible; Examinee may mistakenly believe that the flow instrument is located at the pump discharge.

D - CORRECT; A header rupture on the RBCCW inlet to the letdown heat exchanger will indicate high flow on the "B" RBCCW header flow instrument. The letdown heat exchanger is downstream of the flow instrument.

References

MP-DWG-000-25203-26022

NO Comments or Question Modification History at this time.

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

Number AK3.04 **RO** 3.5 **SRO** 3.7 **CFR Link** (CFR 41.5,41.10 / 45.6 / 45.13)

Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: Effect on the CCW flow header of a loss of CCW

Question #: **8** | **Question ID:** 67415 | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

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

The plant tripped due to a Small-Break Loss Of Coolant Accident and the following plant conditions presently exist:

- The crew has entered EOP 2532, Loss Of Coolant Accident.
- CETs are at 556°F.
- The US has directed a plant cooldown be commenced per the EOP.

Which of the following plant cooldown requirements apply at this time?

☐ **A** Cooldown at any rate </= 100°F/hr.

☑ **B** Cooldown at a rate > 40°F/hr and </= 100°F/hr.

☐ **C** Cooldown at any rate </= 50°F/hr.

☐ **D** Cooldown at a rate > 30°F/hr and < 60°F/hr.

Question Misc. Info: MP2*LORT EOP, 2532, NRC-2016

Justification

A - WRONG; This is the Tech. Spec. Limit at this temperature, not the required C/D rate as directed by the EOP.
Plausible; Examinee may recall that EOP 2537 directs a C/D at the maximum attainable rate (not to exceed TS limits).

B - CORRECT; EOP 2532 Notes proceeding the step that directs the plant cooldown (step 17) states that the rate is > 40°F/hr not to exceed the Tech. Spec. Limit of 100°F/hr.

C - WRONG; RCS temperature at this time does not warrant limiting the C/D to this rate and the EOP requires the rate be > 40°F.
Plausible; Examinee may remember the 50°F/hr Tech. Spec. C/D limit and confuse it with the 40°F C/D requirement.

D - WRONG; 30°F/hr is too low for the existing conditions and EOP guidance.
Plausible; Examinee may recall that EOP 2534 has these C/D requirements based on the concern for uncoupling the SGs.

References

EOP 2532

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.4.20 | **RO** 3.8 | **SRO** 4.3 | **CFR Link** (CFR: 41.10 / 45.13)

Knowledge of the operational implications of EOP warnings, cautions, and notes.

Question #: **9** | **Question ID:** **8000009** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

Rev. **1** | ☑ **Selected for Exam** | **Origin:** **Bank** | ☐ **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 indications, observed less than 20 seconds after the turbine trip, would indicate that the ATWS Mitigation Circuit (i.e.; Diverse SCRAM System) triggered to mitigate the ATWS?

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

☐ **B** Both of the MG Set 480 VAC supply breakers are open.

☑ **C** Both AFW pumps are running and both AFRVs are open.

☐ **D** Both PORVs indicate they opened and closed.

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

Justification

A - WRONG; The TCBs are tripped open, normally, by the RPS, NOT the DSS. This would be "normal" indication that the reactor tripped. Plausible; If the student believes the DSS is an alternate means of tripping the TCBs to shut down the reactor.

B - WRONG; The DSS actuating trips both MG set output contactors as an additional way to shutdown the reactor, separate from RPS. Plausible; If the student remembers the MG set power is removed by the DSS, but not how.

C - CORRECT; Although the load reject would cause a spike in SG pressure and result in a higher than expected shrink in SG level, the AFAS has a time delay to trigger on low SG level of 3 minutes and 25 seconds. However, if the DSS senses a high RCS pressure (>2400 psia) combined with NI control channel power > 20%, the time delay to trigger is reduced to 10 seconds.

D - WRONG; The PORVs are triggered by a high RCS pressure, as seen by the PZR safety channels. The setpoint for the PORV trigger on high pressure is lower than the setpoint for the DSS trigger. A plant trip on load reject would cause a substantial rise in RCS pressure, which could easily result in the PORVs being triggered, regardless of weather the DSS actuated.
Plausible; The RPS setpoint to open the PORVs is within a couple pounds of the DSS trigger value. Therefore, seeing that the PORVs have opened could be construed that the DSS actuated.

References

LP diverse scram DRW

NO Comments or Question Modification History at this time.

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

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

Ability to determine or interpret the following as they apply to a ATWS: System component valve position indications

Question #: **10** | **Question ID:** **2016005** | ☑ **RO** | ☐ **SRO** | ☐ Student Handout? | ☐ Lower Order?

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

The plant tripped from 100% power due to degrading condenser vacuum and the following conditions now exist:

- Steam Generator Tube Rupture of ~120 gpm occurred on the #2 S/G at the time of the trip.
- MS-2A, REHEAT STM SPLY TO MSR 1A, is stuck partially open due to valve binding.
- PZR sprays were being forced pre-trip and the system is still in that alignment.
- Main Condenser back pressure peaked at 15" and is now slowly going down.
- RCS Tcold is at 535°F and stable.
- RCS pressure is at 1700 psia and stable.
- PZR level is at 30% and stable.
- All applicable actions of EOP 2525, Standard Post Trip Actions are being implemented.

Which one of the following actions/conditions, occurring independently, would result in a rise in the RCS leak rate over the subsequent three minutes?

☑ **A** The "B" Charging Pump handswitch is taken out of the "Pull-To-Lock" position.

☐ **B** The forcing of pressurizer sprays is secured and the system is returned to normal.

☐ **C** MS-2A, REHEAT STM SPLY TO MSR 1A becomes unstuck and then fully closes.

☐ **D** Control of the "B" ADV transfers to the Foxboro IA and no adjustments are made.

Question Misc. Info: MP2*LORT CEN-152, SGTR, 2534, NRC-2008, NRC-2016 (8054019) Modified

Justification

A - CORRECT; The RO is required to manually actuate SIAS when RCS pressure drops below 1800 psia, and start all available charging pumps. With a SIAS signal, the "B" charging pump will immediately start when the RO takes handswitch out of P-T-L. PZR level stable at this pressure means the RCS leak rate is at equilibrium with the existing injection flow. Raising injection flow with the "B" charging pump will raise PZR level and therefore, raise RCS pressure and the leak rate.

B - WRONG; RCS pressure is below the setpoint normally used to force spray flow and below the pressure where all backup heaters automatically turn on. Therefore, returning the system to normal will not have an immediate impact on the pressure control auto actions.
Plausible: Examinee may focus on the fact that forcing PZR spray flow will "buffer" any rise in RCS pressure caused by safety injection flow or RCS temperature rise and therefore securing it will cause RCS pressure to rise faster than it already is.

C - WRONG; MS-2A being partially open will result in a slight additional steam demand and heat loss to the RCS. However, the ADVs are set to maintain the RCS at about 535°F, and the conditions given imply the combined heat sinks have reached equilibrium. Closing MS-2A would simply transfer the heat loads to the ADVs, resulting in minimal RCS heatup and pressure rise.
Plausible: Examinee may believe closing MS-2A will result in a measurable RCS heatup and pressure rise, causing a rise in RCS leakage.

D - WRONG; The auto setpoint for the ADVs is 920 psia, which equates to about 535°F Tavg. When control of the valve transfers to the Foxboro IA, the valve setpoint is automatically set to 1200psia, failing the ADV fully closed. This would cuase the pressure in the #2 S/G to rise, which would also raise RCS temperature. However, the other S/G"s SDV is operating in auto just fine and will immediately open further to buffer the RCS temperature rise, but not have as large an impact on the rise in #2 S/G pressure. Therefore, this action will effectively raise SG pressure more than RCS pressure, which would lower the leak rate.
Plausible; Examinee may believe that reducing steam demand will raise RCS temperature, causing a rise in the RCS/SG delta-P and a rise in the leak rate.

References

OP 2304E

NO Comments or Question Modification History at this time.

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

Number EK1.02 **RO** 3.2 **SRO** 3.5 **CFR Link** (CFR 41.8 / 41.10 / 45.3)

Knowledge of the operational implications of the following concepts as they apply to the SGTR: Leak rate vs. pressure drop

Question #: **11** **Question ID:** **54383** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☐ **Lower Order?**

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

The following plant conditions exist:

- The plant tripped from 100% power due to an Excess Steam Demand Event
- The #1 Steam Generator (S/G) is determined to be the faulted steam generator.
- VA-10 has FAILED (deenergized) at the time of the trip.
- The Unit Supervisor has ordered all feed secured to the #1 S/G.
- The BOP has noted that Fac. 2 Auto Aux. Feedwater has just ACTUATED.

Which one of the following is required to secure Auxiliary Feedwater flow to the #1 S/G?

☐ **A** Place BOTH S/G Auto Aux Feedwater Override Switches in the 'Pull-To-Lock' position and set the #1 SG Aux Feed Reg. Valve controller to Manual-Close.

☐ **B** Place the #1 S/G Aux Feed Reg Valve RESET-NORM-OVRD switch in the 'Override' position and set the Aux Feed Reg. Valve controller to Manual-Close.

☑ **C** Take Local-Manual control of the #1 S/G Aux Feed Reg. Valve and close it.

☐ **D** Secure the 'A' Aux Feedwater pump and close FW-44, Aux Feed Water X-Tie.

Question Misc. Info: MP2*LORT*6615 AFW-01-C, AFW, 2322, 2536, AFAS, NRC-2016

Justification

A - WRONG; With VA-10 deenergized, the #1 AFRV cannot be closed from the control room.
Plausible; Examinee may consider this as it is the procedure directed action for securing AFW to a ruptured SG.

B - WRONG; The controller to the #1 AFRV is deenergized, so the override function will not work.
Plausible; Examinee may recall that this action is used when only one SG needs to be manually overridden.

C - CORRECT; This will stop flow to the #1 S/G due to the loss of VA-10. This is because VA-10 powers all Fac. 1 AFAS, which requires power to perform any remote functions. However, Fac. 2 AFAS still has power and will deenergize the DC solenoids for BOTH Aux. Feed Reg. valves, causing them to go full open.

D - WRONG; The 'A' AFW pump would not be running because VA-10 powers the circuit that would have started it.
Plausible; Examinee may recall that this action is taken if a loss of DC control power has occurred for the affected facility after AFAS has triggered, rather than a loss of AC control power.

References

AOP 2501

NO Comments or Question Modification History at this time.

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

Number EA1.2 **RO** 3.5 **SRO** 3.9 **CFR Link** (CFR: 41.7 / 45.5 / 45.6)

Ability to operate and/or monitor operating behavior characteristics of the facility as they apply to the Excess Steam Demand.

Question #: **12** **Question ID: 8080324** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☑ **Lower Order?**

Rev. 0 ☑ **Selected for Exam** **Origin: Bank** ☐ **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.

Question Misc. Info: MP2*LOIT, NRC, 2530, Battery, 125 VDC, NRC-2008, NRC-2016

Justification

A - WRONG; 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.
Plausible; Examinee may focus on the requirement to start a charging pump within 60 minutes to maintain RCS Inventory Safety Function, which could impact RCS HR.

B - WRONG; 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.
Plausible; Examinee may focus on the requirement to start a charging pump within 60 minutes is for boric acid injection to maintain the Reactivity Control Safety Function.

C - WRONG; DC power is not lost during a station blackout; therefore, this is not a concern.
Plausible; Examinee may recall the 60 minute limit is based on loss of DC power, but not the specifics of the reason.

D - CORRECT; EOP 2530, Step 2.22, 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.

References

EOP 2530

NO Comments or Question Modification History at this time.

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

Number EA1.05 **RO** 3.3 **SRO** 3.6 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

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

Question #: **13** | **Question ID:** **1000021** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The plant is at 100% power, steady state, with 480 V buses 22A and 22B cross-tied due to 22A's 4160/480 VAC step-down transformer being tagged out.

Based on the difference between RCS and pressurizer boron concentration, the crew has commenced forcing pressurizer sprays.

The plant then trips and 24D does not transfer to the RSST. All other plant systems and components function as designed.

What effect would this have on the pressurizer heaters, prior to any operator post-trip actions?

☑ **A** Groups '1' and '2' backup heaters in 'Pull-To-Lock'.
Groups '3' and '4' backup heaters in 'Normal-After Close'.
Groups '1', '2' and '4' backup heaters de-energized.

☐ **B** Groups '1' and '3' backup heaters in 'Pull-To-Lock'.
Groups '2' and '4' backup heaters in 'Normal-After Close'.
All Groups of backup heaters de-energized.

☐ **C** Group '1' backup heaters in 'Pull-To-Lock'.
Groups '2', '3' and '4' backup heaters in 'Normal-After Close'.
Groups '1', '2' and '4' backup heaters de-energized.

☐ **D** Groups '1' and '2' backup heaters in 'Normal-After Close'.
Groups '3' and '4' backup heaters in 'Normal-After Close'.
Groups '1' and '2' backup heaters de-energized.

Question Misc. Info: MP2*LOIT, 2654B, pzr heaters, 480V, 2344A, MB-05632, NRC-2008 (K/A: 010-K2.01), NRC-2016

Justification

A - CORRECT; OP 2204, Section 4.2 , has guidance for energizing heaters as required and adjusting the in service pressure controller to maintain pressure. Cross-tying 480 V buses requires that both associated backup heater groups be tagged out due to the potential to overload of 22B supply transformer; therefore, only group 3 and Group 4 heaters are available for forcing sprays. On a loss of power that affects 24D, 24B would be de-energized and thus de-energize B/U Htr Gp '4'. However, B/U Grp '3' is energized from 22C via 24A, which would normally be powered by the RSST post-trip.

B - WRONG; Group 1 and Group 2 heaters are NOT available. With Bus 22B supplying Bus 22A, the heaters are placed in Pull-To-Lock and tagged to prevent overloading Bus 22B transformer.
Plausible; Examinee may swap the heater groups that are required to be removed from service on X-tie of busses based on the assumption that the heater groups are divided between facility 1 and 2 in the same manner as other plant components.

C - WRONG; Placing Group 2 heaters in Pull-To-Lock will help to prevent overloading Bus 22B by keeping loads on 22B at a minimum, but the procedure for cross tying 480 Volt buses does NOT allow energizing backup heaters on Bus 22A. The limit is imposed too protect Bus 22B transformer, not the X-tie breaker.
Plausible; Examinee may recall the power limit affect on X-tie, but not the specific component affected, leading them to believe that only one group of B/U heaters needs to be removed from service on the X-tied busses.

D - WRONG; The limiting factor in bus cross-tie is still applicable, even during normal bus loading.
Plausible; Examinee may believe the heaters are only tagged out if not needed for a specific plant evolution as the overload concern is based on engineering analysis assuming "worse case" bus power demands.

References

OP 2344A

Comments and Question Modification History

K/A not match

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

Number AA2.17 **RO** 3.4 **SRO** 3.6 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Operational status of PZR backup heaters

Question #: **14** | **Question ID:** **5000009** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

Power level is at 92% when numerous control board annunciators alarm and the BOP reports all three white indicating lights for #2 SG Feedwater Control are out.

Based on the applicable AOP, which of the following actions will the BOP take and why?

☐ **A** Press RX TRIP TCBS pushbuttons and close #2 SG FRV Blocking Valve FW-42B to prevent overfill.

☑ **B** Press 'A' or 'B' SGFP MAN pushbuttons and control pump speed manually to maintain level in #2 SG.

☐ **C** Press LIC-5269, #2 SG FRV Controller MAN pushbutton and to maintain #2 SG level within the desired operating band.

☐ **D** Press #2 SG FRV DOWNCOMER RESET pushbutton and control valve in manual to restore level to between 60 and 75%.

Question Misc. Info: MP2 (FWC-01-C RO-8A), NRC-2005 (mod stem to BOP quest.), NRC-2016

Justification

Indications caused by a loss of Vital Instrument Bus VA-20. This bus supplies control power to #2 SG FRV. Loss of power will cause normally open solenoid valves to close on the air supply lines to the Main and Bypass FRVs, which will fail in "as-is" position. Each FRV has three normally lit white control status lights. They indicate low instrument air header pressure, high or low controller output or loss of control power. All three lights will extinguish if power Bus VA-20 is lost.

CHOICE (A) - NO WRONG: The ARP directs the operator to maintain power level constant. Given the indications and the slow rate of power decrease, the operator will be able to control SG level by varying SGFP speed.
VALID DISTRACTOR: applicant may assume that the loss of control while conducting a downpower will require a manual reactor trip. If the reactor is tripped, action to isolate feedwater would be appropriate since the FRV will not close. The AOP for loss of the bus (2504D) contains a caution that warns operator the FRV will not close if a reactor trip occurs and states that since FRV fails 'as is', a SG level transient may occur. This caution prepares operator for one possible outcome, but the following procedural guidance makes it apparent that actions are available to control SG level, thereby avoiding the need for a reactor trip.

CHOICE (B) - YES ARP directs operator, if necessary, to place both SGFPs in manual and to control level by pump speed. This action is necessary since without control power, the FRV cannot be remotely positioned.

CHOICE (C) - NO WRONG: With a loss of control power, the FRV cannot be controlled remotely in auto or in manual.
VALID DISTRACTOR: the alarm response, written for multiple possible causes for a FRV lock condition, does direct manual control of the FRV. The applicant must correctly diagnose the cause of the problem based on given indications in order to determine the correct ARP actions.

CHOICE (D) - NO WRONG: Downcomer reset will not affect FRV control until control power is restored.
VALID DISTRACTOR: the ARP, written for multiple possible causes for a FRV lock condition, does direct the operator to press the pushbutton to restore manual control. The applicant must correctly diagnose the cause of the problem based on given indications in order to determine the correct ARP actions.

References

ARP 2590D-030

NO Comments or Question Modification History at this time.

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

Number AK3.01 **RO** 4.1 **SRO** 4.4 **CFR Link** (CFR 41.5,41.10 / 45.6 / 45.13)

Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus

Question #: **15** **Question ID:** **55206** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☑ **Lower Order?**

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

A loss of 125 VDC Bus 201A causes a plant trip. Buses 25A and 24C fail to transfer to the RSST.

The 'A' D/G will ______________________.

☐ **A** remain shutdown and CANNOT be started from the Main Control Board C08.

☐ **B** start and run at 900 rpm from the Digital Reference Unit with low lube oil pressure and overspeed trips.

☐ **C** come up to speed on the electrical governor and automatically load Bus 24C.

☑ **D** come up to speed on the mechanical governor and have only limited protective features available.

Question Misc. Info: MP2*NLO*1476 [063 LVD-04-C 4173] (6/3/97) 2345, CWT, (LVD-00-C RO-15), NRC-2005, NRC-2016

Justification

CHOICE (A) - NO WRONG: The diesel will start and run on the mechanical governor.
VALID DISTRACTOR: Applicant may think diesel will remain shutdown.

CHOICE (B) - NO WRONG: The diesel will start and run on the mechanical governor. The only available protective feature on a loss of DC control power is mechanical overspeed.
VALID DISTRACTOR: Applicant may think the low oil pressure trip will be available due to an emergency start challenged and because the precaution in the operating procedure may think that the DRU is controlling the EDG.

CHOICE (C) - NO WRONG: The diesel will start and run on the mechanical governor and will not automatically load bus.
VALID DISTRACTOR: Diesel is designed to auto start and auto load on a loss of power to Bus 24C. Applicant may think diesel will function as designed.

CHOICE (D) - YES The diesel generator air start solenoid valves fail open on a loss of DC. The diesel will start and run on the mechanical governor with only the overspeed trip available; all other trips need DC to operate. The diesel output breaker will not close without DC control power, so the diesel can not provide power to the bus. Question requires applicant to understand effects of loss of DC on the EDG and on Bus 24C.

References

AOP 2505A

Comments and Question Modification History

Changed "B" distracter to make it more plausible and modified "A" to make it longer.djj

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

Number AK3.01 **RO** 3.4* **SRO** 3.7 **CFR Link** (CFR 41.5,41.10 / 45.6 / 45.1)

Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Use of dc control power by D/Gs

Question #: **16** | **Question ID: 2016040** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The plant is operating at 100% power during the summer months, when an electrical malfunction causes Bus 24A, NSST supply breaker (A102) to trip.

All systems respond as expected.

The Crew has entered the following procedures:
- AOP 2502A "Loss of Non-Vital 4.16 kV Bus 24A"
- AOP 2502C "Loss of Vital 4.16 kV Bus 24C"

Which of the following actions are procedurally required to be performed and why?

☑ **A** Manually over-ride and open 2-SW-3.2B to ensure adequate cooling to the TBCCW System.

☐ **B** Secure Charging and Letdown to secure Emergency Boration, stopping the Reactor down power.

☐ **C** Cross-tie Buses 22A and 22C to 22B and 22D to ensure all available TBCCW pumps can operate.

☐ **D** Ensure the East DC Swg. Room Vital Chiller X-169A is in service to maintain temperature < 97°F.

Question Misc. Info: MP2*LOIT AOP, 2502, SW, TBCCW, NRC2016

Justification

A - CORRECT; Procedural guidance per AOP 2502C requires 2-SW-3.2B over-ridden and opened, which is the "A" SW isolation to the "C" TBCCW heat exchanger to ensure adequate SW cooling restored to TBCCW to allow for continued Plant Operations. During the summer months this action is required in a timely manner due to the heat loads on the system.

B - WRONG; Although the Boric Acid Storage Tank gravity feed valves to the suction of the Charging pumps open on an EDG LNP start signal the back pressure from the VCT prevents any Boric Acid from getting to the suction of the Charging pumps.
Plausible; The Operator may think that the Charging pump suction is coming from the BAST which is correct, through the Gravity Feed valves, however if this were Z2 facility then the BAST pumps would start on the LNP and with a discharge pressure greater than the VCT this would cause a rapid down power of the Reactor.

C - WRONG; Procedural guidance requires cross tying Non-Vital 480 VAC buses to ensure proper Circ Water alignment to maintain condenser vacuum and not TBCCW flow using all three pumps.
Plausible; During the summer months for plant operations the TBCCW system occasionally operates all 3 TBCCW pumps due to the higher flow rates required to maintain components cooled by TBCCW therefore an Operator may think that having 3 pumps operating is a requirement for continued plant operations, also "TBCCW HDR PRESS LO" alarm will actuate due to the loss of cooling, thereby causing TCV's to open reducing system pressure ARP requires starting additional TBCCW pumps.

D - WRONG; Although Vital Chiller X-169A may start depending on the Chill Water System configuration the most limiting TSAS is 2 hours for a NON OPERABLE vital switch gear cooling. Maintaining DC Switch gear room temperature is not referred to in the loss of 24C procedure, because it is assumed to have automatically started, only requiring a check of SWGR temperature if a Board alarm is received.
Plausible; Examinee may assume the loss of cooling to the non-vital swgr room chillers will require verification of the vital chiller operation, as loss of cooling to these rooms will impact the continued availability of the Vital Instrument AC busses.

References

AOP 2502C

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 062 Loss of Nuclear Service Water

Number AA1.07 **RO** 2.9 **SRO** 3.0 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the Loss of Nuclear Service Water (SWS): Flow rates to the components and systems that are serviced by the SWS; interactions among the components

Question #: **17** | **Question ID: 3100002** | ☑ **RO** | ☐ **SRO** | ☐ Student Handout? | ☑ Lower Order?

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

The plant is operating at 100% power when the Balance of Plant (BOP) operator reports that Instrument Air header pressure is at 95 psig and lowering.
Immediately following, the Turbine Building PEO reports a large unisolable leak just downstream of the "D" Instrument Air Dryer After Filters.

Assuming Instrument Air header pressure continues to lower, at what pressure in the Instrument Air System is a plant trip or shutdown required (by procedure) and why?

☑ **A** When pressure lowers to < 80 psig.
The loss of many important controls, such as Feedwater, could degrade plant conditions at the time of the trip; therefore, the reactor must be tripped when control of important systems could become challenged.

☐ **B** When pressure lowers to < 80 psig.
At < 80 psig the Instrument Air/Station Air Crosstie valve opens. Continued operation with Station Air supplied to valves and controllers will result in erratic operation of components due to the higher moisture content of Station Air.

☐ **C** When pressure lowers to < 85 psig.
The loss of many important controls, such as Feedwater, could degrade plant conditions at the time of the trip; therefore, the reactor must be tripped when control of important systems could become challenged.

☐ **D** When pressure lowers to < 85 psig.
At < 85 psig the Instrument Air/Station Air Crosstie valve opens. Continued operation with Station Air supplied to valves and controllers will result in erratic operation of components due to the higher moisture content of Station Air.

Question Misc. Info: MP2*LOUT, IA, OP 2332B, AOP 2563, MB-04688, MB-04688, NRC-2003, NRC-2009 (SRO Question), NRC-2016

Justification

A - CORRECT; AOP 2563, Discussion section 1.2. When IA pressure lowers to less than 80 psig, the Feed Regulating Valves may lock up resulting in over feeding of Steam Generators after the trip. Additionally, the Steam Dumps may not open resulting in opening of the Main Steam Safeties as the only initial means of removing decay heat.

B - WRONG; Although it is less desirable to operate with IA cross tied with SA, there are NO restrictions; therefore, NO requirements to trip or shut down the plant.
Plausible; Examinee may feel that continued operation with Instrument Air supplied by Station Air is NOT allowed for extended periods of time because the SA compressor does not have an air dryer and the higher moisture content in the air may affect vital plant component operation.

C - WRONG; The plant trip requirement is at < 80 psig, not 85 psig.
Plausible; Examinee may confuse the pressure where IA & SA automatically X-tie with the low pressure trip value, or feel that the potential for oil contamination of certain plant components is not allowed.

D - WRONG; Although the Instrument Air/Station Air Crosstie valve automatically opens at ~ 85 psig, continued operation with Station Air cross tied to Instrument Air is acceptable.
Plausible; Examinee may remember when the Station Air Cross Tie valve opens and believe that continued operation with Station Air supplying Instrument Air is NOT allowed, because the SA compressor does not have an air dryer (i.e.; higher moisture content).

References

AOP 2563

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.1.20 **RO** 4.6 **SRO** 4.6 **CFR Link** (CFR: 41.10 / 43.5 / 45.12)
Ability to interpret and execute procedure steps.

Question #: **18** **Question ID:** **80874** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☑ **Lower Order?**

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

The plant is operating at 100% power and a "Degraded Voltage" condition exists.

Due to the existing conditions, AOP 2580, Degraded Voltage, requires a plant trip.

Which one of the following statements describes actions that must be carried out per AOP 2580, Degraded Voltage, prior to tripping the plant and why?

☑ **A** Open and disable the RSST breakers on all busses to prevent them from closing and energizing plant equipment at a degraded voltage.

☐ **B** Throttle the discharge valves to the standby Service Water and RBCCW Pumps, to ensure availability on the potential Loss Of Offsite Power.

☐ **C** Start "A" and "B" Diesel Generators and place them in service on bus 24C and 24D, to protect safety equipment from the degraded voltage post-trip.

☐ **D** Align bus 24E to get power from the Unit Three Station Blackout Diesel Generator, to ensure a power source is available on the potential Loss Of Offsite Power.

Question Misc. Info: MP2*NLIT/LOIT 2580, Drgraded Voltage, Required Actions, NRC-2016

Justification

A - CORRECT; In order to prevent transferring to the RSST, which is getting power from a degraded grid voltage, the RSST breakers must be disabled so they will NOT close on the impending plant trip.

B - WRONG; The procedure does not direct any action on the standby equipment until they are actually needed. This would be wasted personnel resources at a time when they are needed elsewhere.
Plausible; Student may recognized the potential damage a degraded voltage condition can have of vital equipment and postulate a need to have the standby equipment at the ready.

C - WRONG; This would put the EDGs in a very vulnerable position in that they would attempt to pick up the whole grid when voltage dropped.
Plausible; Student may believe having the EDGs already on line eliminates the risk of them failing to start and load when they are truly needed.

D - WRONG; This would be another waste of personnel resources and is not directed by the AOP. When, and if, it becomes necessary, there should be time to take this action.
Plausible; Student may believe the probable total loss of offsite power post-trip increases the potential risk of a Station Blackout, thereby warranting the early action to get a bus ready to be picked up by the SBO diesel.

References

AOP 2580

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

NRC K/A Generic **System** 2.4 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 #: **19** | **Question ID:** **2016008** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**
Rev. **0** | ☑ **Selected for Exam** | **Origin:** **New** | ☐ **Past NRC Exam?**

Plant power ascension is in progress from 60% power to 100% power.

The Reactor Operator continued to withdrawal Rods during the power ascension.

Which of the following Plant indications would indicate that a Group 7 rod became stuck on a Rod withdrawal but separated from the Extension shaft?

☑ **A** C04 AA-08 “RX POWER ΔT CH DEVIATION” and BA-12 “NIS CHANNEL DEVIATION HI” on a core perimeter CEA.

☐ **B** CEAPDS CMI on Group 7 Deviation when the stuck CEA became more than 8 steps lower than the Group.

☐ **C** RPS PRE TRIP on TM/LP due to the abnormal ASI input to the Thermal Low Pressure Calculator.

☐ **D** C04 BB-12 “PWR RATIO HI/LO” due to a deviation between Safety Channel and Control Channel NI's.

Question Misc. Info: MP2*LOIT CED-01-C, CEDS, TS, FRT, TQ, NRC-2016

Justification

A - CORRECT; These Alarms are the expected response for a Group 7 rod stuck at the perimeter causing a difference in NIS power between the various detectors.

B - WRONG; Extension Shaft provides all the Reed indications to CEAPDs which would continue to move.
Plausible; The Examinee would pick this answer to be correct if the Rod and the Extension shaft did not separate.

C - WRONG; Axial power is minor affect by a Rod out of step, but the major affect would be radial distribution.
Plausible; Examinee may assume the stuck Rod would affect axial distribution which has an effect on the TM/LP setpoint.

D - WRONG; Axial power is minor affect by a Rod out of step, but the major affect would be radial distribution.
Plausible; Examinee may assume the stuck Rod would affect axial distribution enough to trigger C04 BB-12 “PWR RATIO HI/LO” alarm.

References

AOP 2556, ARP 2590C-057, ARP 2590C-089

NO Comments or Question Modification History at this time.

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

Number AA2.01 **RO** 3.3 **SRO** 4.1 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Inoperable / Stuck Control Rod: Stuck or inoperable rod from in-core and ex-core NIS, in-core or loop temperature measurements

Question #: **20** **Question ID: 2000005** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☑ **Lower Order?**

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

Why are the 'Steam Dump & Bypass Valves' inhibited from opening on loss of condenser vacuum?

☐ **A** Ensures the LP turbine is not operated with high windage losses to prevent over heating of the blade tips.

☐ **B** Ensures that the Circulating Water NPDES discharge temperature limit of 105 °F is not exceeded.

☑ **C** Ensures the LP turbine exhaust hoods and condenser are not over-pressurized when circulating water is lost.

☐ **D** Ensures that CPF resins are not exposed to condensate temperatures above the recommended maximum.

Question Misc. Info: MP2*LOIT/LOUT, MSS, SD&BV, 2574, NRC-2016

Justification

A -WRONG; The CDV interlock is 10" Hg but the turbine trip setpoint is 7.5" Hg, which is designed to prevent turbine operation with high backpressure.
Plausible; Examinee may focus on the real concept of vacuum loss causing high blade tip temperatures.

B -WRONG; The vacuum inhibit is set for 15" Hg backpressure, which equates to a Tsat of ~180°F.
Plausible; Examinee may consider the problem of high discharge temperatures that occurs when the cause of degraded vacuum is a reduction in Circ. Water flow, and adding more steam flow would make it worse.

C: CORRECT; The SD&BV inhibit in conjunction with the LP turbine relief diaphragms ensure that the exhaust hoods and condenser are not over-pressurized;

D -WRONG; Typical max temperatures for resins is ~145°F.
Plausible; Examinee may equate the loss of condensate depression as an impact on demin resin, which it could be if condensate temperature where allowed to get that high.

References

LP_MT-00-C_R6_Pg15-16

Comments and Question Modification History

Choice "A" changed from "Ensuring adequate NPSH for the condensate pumps requires a maximum Psat based on a maximum Tsat." to existing wording, based on Validator feedback. - RLC

NRC K/A System/E/A **System** 051 Loss of Condenser Vacuum

Number AK3.01 **RO** 2.8* **SRO** 3.1* **CFR Link** (CFR 41.5,41.10 / 45.6 / 45.13)

Knowledge of the reasons for the following responses as they apply to the Loss of Condenser Vacuum: Loss of steam dump capability upon loss of condenser vacuum

Question #: **21** **Question ID: 5000047** ☑ RO ☐ SRO ☐ Student Handout? ☐ Lower Order?

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

Refueling is in progress on Unit 2. During normal rounds, the Aux Building PEO reports that the red light on the SFP SW Area Radiation Monitor (RM-8139) local module is illuminated.

Which of the following would explain the condition, and the reason for the required action, per the applicable procedure?

☐ **A** Power to the RM-8139 was restored and requires the ESF bistable be reset, to allow the rad monitor to perform its design function of triggering AEAS.

☐ **B** Local horn silence key switch is in the OFF position and must be returned to normal, to ensure the rad monitor can warn of a local high radiation condition.

☑ **C** A high radiation condition in spent fuel pool area is indicated, and HP must be notified to perform area surveys to verify actual spent fuel pool radiation levels.

☐ **D** "FUEL AREA RADN AEAS" switch for RM-8139 at ESF sensor cab is in INHIBIT and must be returned to OPER to allow the rad monitor to function as designed.

Question Misc. Info: MP2*LOIT Rad. Mon., SRO, NRC-2005 [K/A 061, ARM, AA2.01], NRC-2011, 55.43(b)(4), NRC-2016

Justification

CHOICE (A) - NO WRONG: Loss of power extinguishes the light and with the bistable tripped it does not effect the local light.
VALID DISTRACTOR: Plausible that function of red light is to indicate a loss of power.

CHOICE (B) - NO WRONG: Local horn silence keyswitch disables the audible alarm but leaves the red light lit.
VALID DISTRACTOR: Plausible that function of red light is to inform that audible is defeated.

CHOICE (C) - YES Local red light illuminates on sensed high radiation condition at a reading exceeding 50mR/hr.

CHOICE (D) - NO WRONG: Keyswitch at ESF sensor cabinet functions to inhibit the trip and change logic from 2 /4 to 2 out of 3.
VALID DISTRACTOR: Plausible that red light designed to provide local indication of defeated input to ESAS.

References

ARP2590H-015, RMS-00-C

Comments and Question Modification History

Original question #54838 was replaced due to conflicts with other questions on the exam.
Quesiton and all choices enhanced to fit the chosen K/A and raise the LOD. - RLC

NRC K/A System/E/A **System** 061 Area Radiation Monitoring (ARM) System Alarms

Number AK3.02 **RO** 3.4 **SRO** 3.6 **CFR Link** (CFR 41.5,41.10 / 45.6 / 45.13)

Knowledge of the reasons for the following responses as they apply to the Area Radiation Monitoring (ARM) System Alarms: Guidance contained in alarm response for ARM system

Question #: **22** **Question ID: 2016007** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☑ **Lower Order?**
Rev. 0 ☑ **Selected for Exam** **Origin: Mod** ☐ **Past NRC Exam?**

The control room has been evacuated due to an Appendix "R" fire. The reactor has been tripped and all switches on the Bottle-Up Panels are in the "ISOL" position.

Per the applicable Appendix "R" Fire Procedure, which one of the following describes additional actions that are expected to be performed within the next fifteen (15) minutes?

☐ **A** 1. Shift the #2 ADV controller on C-21 to Manual mode.
2. Operate the #2 ADV from C-21 to control RCS temperature.

☐ **B** 1. Shift both ADV controllers on C-21 to Manual mode.
2. Operate both ADVs from C-21 to control RCS temperature.

☑ **C** 1. Place all switches on the C-10 panel to the "LOCAL" position.
2. Operate the #2 ADV from C-10 to control RCS temperature.

☐ **D** 1. Place all switches on the C-10 panel to the "LOCAL" position.
2. Operate both ADVs from C-10 to control RCS temperature.

Question Misc. Info: MP2*LOUT, Fire, 2579A, MB-04722, NRC-2016

Justification

A - WRONG; C21 Panel is the preferred "go to" panel when the control room must be evacuated because it allows for better plant control than C-10. However, C-21 cannot be used for an evacuation driven by an Appendix "R" fire because of the need to isolate various control systems from the possibility of a hot short operation.
Plausible; Examinee may note that C-21 is the preferred panel to operate from when the control room has to be evacuated.

B - WRONG; The need to isolate critical control systems during an Appendix "R" fire cause the loss of all but a few Facility 2 powered components. Therefore, only the #2 ADV can be operated remotely and only the #2 SG is aligned to Aux. Feed.
Plausible; Examinee may note that C-21 has the capability to control both ADVs and feed to both SGs, which would be the preferred method of plant control (if the procedure allowed).

C - CORRECT; AOP 2579A lists the required actions, which are: 1. Place all switches on the Bottle-Up Panels in the "ISOL" position; 2. Ensure the reactor is tripped and evacuate the Control Room. 3. (Establish communications using radios) Place all switches on C-10 in the LOCAL position. 4. Place #2 ADV controller in Manual/Close. 5. Perform various electrical alignments and establish feed to #2 Steam Generator with the Terry Turbine.

D - WRONG; Because the normal control circuits cannot be trusted during several types of Appendix "R" fires, control power is isolated to all Facility 1 components and all but a few Facility 2 components. As such, the C-10 panel can only control the #2 ADV.
Plausible; Examinee may recognize that C-21 allows the control of heat removal on both RCS loops, which is especially important considering the added complexity of natural circulation flow in the RCS.

References

AOP 2579A

NO Comments or Question Modification History at this time.

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

Number AA1.01 **RO** 4.3 **SRO** 4.5 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the Control Room Evacuation: S/G atmospheric relief valve

Question #: **23** **Question ID:** **2016009** ☑ **RO** ☐ **SRO** ☐ Student Handout? ☐ Lower Order?
Rev. 1 ☑ Selected for Exam **Origin:** **Mod** ☐ Past NRC Exam?

The plant is in Mode 1 and a containment entry was made to investigate an unidentified RCS leak of approximately 0.8 gpm.

While exiting CTMT, the door interlock mechanism fails, preventing the inner and outer doors from fully closing.

Various ventilation radiation monitors immediately begin to slowly rise.

Containment pressure has slowly risen to 1.0 psig and the ventillation radiation monitors continue to rise.

Which one of the following procedural actions will mitigate the radiation leakage through the containment air lock to the environment?

☐ **A** Start a fourth CAR fan in "Fast" to maximize containment cooling and reduce pressure.

☑ **B** Vent Contaiment through the Hydrogen Purge Flow Path to Enclosure Building Purge.

☐ **C** Verify All Auxiliary Building doors are closed to ensure CTMT INTEGRITY is maintained.

☐ **D** Start both Post Incident Reciculation fans to increase the efficency of the CAR fans.

Question Misc. Info: LOIT*MP2 CTMT, Air Lock, Rad. Release, NRC-2008 (Parent), NRC-2016

Justification

A - WRONG; Four CAR fans cannot be running in fast per OP 2313A.
Plausible; Examinee may refer to Control Room daily Surveillance for CTMT temperature which state START CAR fans as necessary.

B - CORRECT; Control Room Daily Surveillance requires vent Containment due to pressure exceeding the Tech. Spec. For pressure which reduces the D/P reducing the leak rate from CTMT and filtering the effluent through the EBFS filters..

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 Aux. Bldg or the Enclosure Bldg.
Plausible; Examinee may believe this will fix the problem because MP2 CTMT is expected to have some leakage in an accident by design, which is why EBFAS and Enclosure Building integrity (i.e.; Aux. Building doors) are required.

D - WRONG; This would increase the Circulation of air in CTMT but the Tech. Spec. action needs to be addressed for operations.
Plausible; Examinee knows that uniform mixing of the Containment post-incident atmosphere is provided by the Post-Incident Recirculation System allowing the PIR fans which are designed to operate in the air-steam mixture, helping in reducing the temperature and pressure in CTMT.

References

OP 2314B, T.S. 3.6.1.4

NO Comments or Question Modification History at this time.

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

Number AK1.01 **RO** 2.6 **SRO** 3.1 **CFR Link** (CFR 41.8 / 41.10 / 45.3)

Knowledge of the operational implications of the following concepts as they apply to Loss of Containment Integrity: Effect of pressure on leak rate

Question #: **24** **Question ID:** **2016010** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☐ **Lower Order?**

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

The plant tripped due to failure of the "C" RCP breaker. On the trip, busses 24C and 24D failed to transfer to the RSST and are being powered by their respective Emergency Diesel Generators.

All other systems and components are operating as designed for the given conditions.

The crew has entered EOP 2526, Reactor Trip Recovery, and is maintaining the plant in Mode 3.

Which of the following conditions (taken individually) would require reference to another AOP/EOP to maintain the existing core heat removal?

☐ **A** Loss of the only available Condensate Pump.

☐ **B** Inadvertent Main Steam Isolation Signal on Facility 1 only.

☑ **C** The main feeder breaker on VA-20 fails open.

☐ **D** Over current trip of the RSST feeder breaker to Bus 25A.

Question Misc. Info: MP2*LOIT EOP, 2525, ESAS, SGTR, HR, NRC-2016

Justification

A - WRONG; The LNP on the plant trip means the Main Feed Pumps are unavailable and AFW would be required to feed the S/Gs. Therefore, the loss of a running condensate pump would have no impact.
Plausible; Examinee may regard the running condensate pump as a source of feed water to the S/Gs, as it normally would be if the condenser wasn't lost with the LNP.

B -WRONG; Spurious MSI is addressed by AOP-2571, "Inadvertent Emergency Core Cooling System Initiation", which provides direction for maintaining power operation while addressing the problems of inadvertent isolation. However, under the present plant conditions, the components and systems being used for heat removal (Aux. Feedwater and the ADVs) do not receive an MSI signal.
Plausible; The examinee may focus on the MSI signal that closes the MSIVs and trips the main feed pumps.

C - CORRECT; The Loss of Normal Power (LNP) to the vital 4160V AC busses would cause a loss of all Circulating Water Pumps. This would require the operators to close the MSIV, isolating steam to the turbine building, and then break condenser vacuum. The loss of the main condenser means the Main Feed Pumps are unavailable and AFW would be required to feed the S/Gs. During the performance of EOP post trip actions, the BOP would have adjusted the ADVs to control RCS temperature, manually started feeding the S/Gs with Aux Feedwater and taken manual control of the Aux. Feed Reg. Valves (AFRVs) to control S/G level. Loss of VA-20 will cause the "B" Atmospheric Dump Valve (ADV) to fail fully closed and the "B" AFRV to fail full open, as if the valve received the normal control signal for automatic system actuation (AFAS). This would affect the heat removal rate of the #2 S/G, requiring operator action to mitigate. AOP-2504D, Loss of 120V AC Vital Instrument Panel VA-20, would be referenced to mitigate the effects of this power loss. It would require the operators to control the "B" AFRV and the "B" ADV locally using a PEO at the valves.

D - WRONG; This would cause the loss of the "A" RCP and two condensate pumps. The RCPs are required to be operated in specific combinations, depending on RCS pressure, and never alone. However, "B" and "D" RCPs are still powered and are an approved combination of operating RCPs. Therefore, the power loss will have no impact on RCS flow (core heat removal).
Plausible; The Examinee may believe the loss of 25A will take out one of the running RCPs and the remaining two RCPs do not make up an approved combination. This would have required the trip of the remaining RCPs and the transition to EOP-2528, Loss Of Offsite Power/ Loss Of Forced Circulation to mitigate.

References

AOP 2504D

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.4.2 **RO** 4.5 **SRO** 4.6 **CFR Link** (CFR: 41.7 / 45.7 / 45.8)

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

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

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

Question Misc. Info: MP2*LOIT, RCS, 2515, Charging, CVCS, activity, NRC-2008, NRC-2016

Justification

A - WRONG; The second letdown valve is only placed in service if low RCS pressure precludes raising letdown flow to the desired amount. No is no event given that would lower RCS pressure to this level.
Plausible; RCS cleanup is normally performed in Mode 5, which would require a second L/D flow control valve be placed in service. Examinee may recall this "normal" practice and assume it is always required.

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.

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.
Plausible; Examinee may recall the required actions of OP-2204 and assume the two requirements are connected.

D - WRONG; The IXs are maintained in series, therefore a second IX in service would not affect the delta-P across a single exchanger.
Plausible; Examinee may think that under the unusual (and concerning) situation of high RCS activity, the IXs could be realigned to improve RCS cleanup.

References

AOP 2511

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 / 45.6 / 45.13)

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: 2016030** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The plant is operating at 100% power with the following conditions:

- 1 charging pump running
- Normal letdown flow
- All other components normally aligned.

Then, the sensing line for the Letdown Back Pressure Controller breaks off, resulting in a small (< 1 gpm) leak in the letdown line at the separated instrument tap.

Which one of the following conditions would result in the next two to three minutes?

☐ **A** Indicated letdown flow will lower to zero gpm,
Actual letdown flow will lower to zero gpm.

☐ **B** Indicated letdown flow will remain constant,
Actual letdown flow will remain constant.

☐ **C** Indicated letdown flow will remain constant,
Actual letdown flow will lower to zero gpm.

☑ **D** Indicated letdown flow will lower to zero gpm,
Actual letdown flow will remain constant.

Question Misc. Info: MP2 LOUT, CVC-01-C, MB-2395, NRC-2016

Justification

The rupture of the Back Pressure Controller sensing line will cause the controller to see zero pressure in the letdown line, which will cause the controller to close the applicable back pressure control valve. This will isolate letdown flow and cause pressure upstream of the valves to rise until the letdown relief valve (just upstream of the L/D H/X) lifts and diverts all letdown flow to the CLRW. Actual letdown flow is controlled by the Letdown Flow Control Valves, which modulate on PZR level and have no feedback from letdown flow. As charging flow has not changed, PZR level would not have changed and, therefore, the flow control valves will not have changed position. As the letdown relief valve is designed to handle maximum letdown flow, it will lift on pressure and effectively maintain actual letdown flow constant. However, indicated letdown flow is sensed down stream of the closed back pressure control valves, and will therefore go to zero gpm.

A; Wrong - Actual letdown flow is a function of PZR level, which would not have changed because charging flow never changed and the leak is not in the RCS.
Plausible; Examinee may consider the effect of the closing back pressure control valves and assume they isolated letdown flow (forgetting about the letdown relief valve opening).

B; Wrong - The rupture will cause the Back Pressure Control valves to close, isolating letdown flow upstream of the indicated letdown flow sensing line, causing indicated letdown flow to go to zero.
Plausible; Examinee may believe letdown flow is sensed upstream of the back pressure control valves and therefore display the actual flow when the relief valve lifts.

C; Wrong - The rupture will cause the Back Pressure Control valves to close, isolating letdown flow upstream of the indicated letdown flow sensing line, causing indicated letdown flow to go to zero.
Plausible; Examinee may remember the affect of the failed sensing line, but confuse where letdown flow is sensed in relation to the back pressure control valves or that the letdown relief valve lifting to rad waste still constitutes "letdown flow" even though it isn't going to the VCT.

D - CORRECT; The rupture will cause the Back Pressure Control valves to close, isolating letdown flow upstream of the indicated letdown flow sensing line, causing indicated letdown flow to go to zero. The back pressure control valve going closed will cause the letdown relief valve to lift, maintaining actual letdown flow (controlled by PZR level) constant.

References

ARP 2590B-059, ARP 2590B-031, CVC-00-C R10-01, LP CONT-01-S

NO Comments or Question Modification History at this time.

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

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

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

Question #: **27** | **Question ID:** 80999 | ☑ **RO** | ☐ **SRO** | ☐ Student Handout? | ☐ Lower Order?

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

The plant was tripped due to a large MSLB on #2 SG inside containment.

The following conditions now exist:

- EOP 2536 ESDE has been entered
- #2 SG has blown dry and was isolated as directed in the EOP.
- RCS temperature and pressure are stabilized.
- All other plant systems and components functioned as designed.

Then, the following changes occur:

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

Which of the following rad monitors would be used to confirm the diagnosis of a SGTR?

☑ **A** Containment Refueling Floor Area radiation monitor.

☐ **B** Containment gaseous or particulate radiation monitors.

☐ **C** Steam Jet Air Ejector or Steam Generator Blowdown rad monitors.

☐ **D** Main Steam Line radiation monitors for either of the ADVs.

Question Misc. Info: MP2*LORT 2540, ESD, SGTR, NRC-2016

Justification

A - CORRECT; Because the steam rupture is in CTMT, the SGTR will simply dump the RCS to CTMT. Although the refueling bridge rad monitor is not at an elevated location inside CTMT, the operation of several CTMT fans, started with the ESD, would substantially assist the mixing of the CTMT atmosphere, and allow this rad monitor to better sense radioactivity from the leaking RCS.

B - WRONG; The ESD would have triggered a SIAS/CIAS actuation, which would isolate the CTMT gaseous and particulate rad. monitor sampling flow paths before the SGTR occurred.
Plausible; Examinee may focus on the rad monitors normally directed to be used to diagnose an RCS leak into CTMT.

C - WRONG; The ESD in CTMT would have triggered an MSI, which closes the MSIVs and isolates the Main Steam header.
Plausible; Examinee may consider these Rad. Monitors as they are mentioned in procedures as the first indication of a SGTL or SGTR.

D - WRONG: The MSL Rad. Monitors are designed to sense a steam release with a SGTR, combined with a fuel failure. Although procedures direct their use as an indication of a SGTR, when coupled with lowering PZR level, this is based on the detection of N-16 leaking from the RCS into the Main Steam header while at power. The RCS would not normally contain enough radiation, post-trip, to trigger these detectors.
Plausible; Examinee may consider the MSL Rad. Monitors based on their proceduralized use in detection of a SGTR.

References

EOP 2536, EOP 2541-APP01

NO Comments or Question Modification History at this time.

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

Number EA1.1 **RO** 4.2 **SRO** 4.0 **CFR Link** (CFR: 41.7 / 45.5 / 45.6)

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

Question #: **28** | **Question ID:** 1100053 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

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

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

Assuming all other systems function as designed, which of the following describes parameter response <u>within the first 30 seconds</u> after the loss of the 6.9 kV buses?

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

Question Misc. Info: MP2*LOIT, RCS, RCP, RPS, NRC-2011, NRC-2016

Justification

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

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

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

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

References

EOP 2528, RCS-00-C

Comments and Question Modification History

Changed Stem from 1 minute to 30 seconds validator comment. Djj

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

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

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

Question #: **29** | **Question ID:** **2016011** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The plant is operating at 100% power, steady state.

VCT pressure is presently 10 psig and being raised by the addition of hydrogen.

The hydrogen addition is not secured in time and the annunciator C02/03 D-7 "VCT PRES HI/LO" alarms.

Which of the following statements describes the effect of the above condition on Reactor Coolant Pump seal bleedoff flow and the applicable procedural action to take?

☐ **A** Lowers to a new stable value due to the rise in VCT pressure decreasing the D/P from the Vapor Seal; refer to OP 2304A Volume Control Portion of CVCS to manually restore RCP bleed off flow.

☐ **B** Lowers to a new stable value when VCT pressure decreases after the completion of the ARP step to place “LTDN DIVERT, CH 500,” in “RWS” (C-02) to restore VCT pressure.

☑ **C** Lowers initially due to the increase in VCT back pressure and then returns to previous due to the Bleedoff pressure controller; refer to OP 2304A Volume Control Portion of CVCS and monitor.

☐ **D** Lowers initially due to the increase in VCT back pressure and then returns to normal after the ARP step to place “LTDN DIVERT, CH 500,” in “RWS” (C-02) to restore VCT pressure.

Question Misc. Info: MP2*LOIT RCS, RCP, Bleedoff, NRC*2016

Justification

"A" WRONG; Bleedoff flow controller uses a pressure setpoint to maintain bleedoff flow which maintains flowrate, any pressure deviation will cause the controller to open or close to restore to the original bleedoff pressure.
Plausible; When placing the system in operation IAW OP 2301C Section 4.1 “Establishing RCP Seal Controlled Bleedoff Flow” the operator manually adjusts the controller output this answer is correct for a controller that is in manual and not for a controller in remote that’s adjusting to pressure fluctuations.

"B" WRONG; This action would be correct if the cause of the high VCT pressure was caused by a high level and not a high pressure which was caused by a hydrogen add as describe in the stem of the question.
Plausible; When CH-500 is diverted to Radwaste the VCT level and pressure will drop the examinee may assume that reducing VCT level and pressure will increase bleedoff flow, although the RCP bleedoff pressure controller will restore bleedoff pressure thereby restoring bleedoff flow to its original value.

"C" CORRECT; In normal, Remote Operation for PIC-215 refer to OP 2304A Attachment 2. Before the controller sees a pressure rise, bleedoff flow will lower until the controller restores bleedoff pressure.

"D" WRONG; RCP Bleedoff flow is restored to normal when the bleedoff pressure controller restores back pressure of the RCP bleedoff.
Plausible; A step in the ARP for VCT high pressure requires placing CH-500 to Divert but only in the case that the high pressure in the VCT was cause due to a high level, the examinee may falsely assume this action to reduce VCT pressure.

References

ARP 2590B-028, OP 2304A

NO Comments or Question Modification History at this time.

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

Number A2.05 **RO** 2.5 **SRO** 2.8 **CFR Link** (CFR: 41.5 / 43.5/ 45.3 / 45/13)

Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effects of VCT pressure on RCP seal leakoff flows

Question #: **30** | **Question ID: 2016032** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The plant is shut down for a refueling outage with the following conditions:

- "A" LPSI Pump and both SDC HXs are in service.
- SDC to RCS (T-351Y) = 150 °F.
- Additional Purification is in operation, with the Purification Ion Exchanger in service, to clean up the RCS.

With NO operator action, which one of the following would adversely affect the RCS cleanup efforts?

☑ **A** The temperature transmitter to TIC-223, LTDN TEMP CNTL, fails low.

☐ **B** VCT level indication, L-226, fails high due to a reference leg malfunction.

☐ **C** Regenerative Heat Exchanger Outlet Temperature Detector, TS-516, fails high.

☐ **D** Pressure transmitter, P-201, to the Letdown Back Pressure Controller, PIC-201, fails high.

Question Misc. Info: MP2*LOIT/LORT, CVCS, Additional Purification, ARP 2590B-033, NRC-2016

Justification

A - CORRECT; With SDC aligned for Additional Purification, some SDC flow is diverted to CVCS letdown, just upstream of the Letdown Heat exchanger. If the outlet temperature detector of the non-regenerative heat exchanger reaches 145 °F, ion exchanger bypass valve (2-CH-520) will shift to the bypass position and bypass flow around the ion exchangers. TIC-223 is the controller for RB-402, which supplies cooling to the Non-Regen HX. The failure will close 2-RB-402, causing a loss of cooling to the Letdown Heat Exchanger, resulting in an increase in temperature into the Purification Ion Exchanger and eventually result in the auto bypass of the ion exchangers.

B - WRONG; This would ordinarily cause the letdown divert valve to send letdown flow from the VCT to rad waste. However, the procedure directs the divert valve be placed in the "VCT" mode, which aligns it to a leg of the system the procedure also isolates. The purification flow path used taps off upstream of the divert valve to the VCT, but if the divert valve changed to "Divert", RCS inventory would be sent to rad waste.
Plausible; Examinee may believe the procedure isolates the normal letdown divert flow path to rad waste to prevent the loss of RCS inventory at this crucial time, but aligns the divert valve to the VCT to ensure a return path to the SDC system.

C - WRONG; A high failure of the Regenerative Hx outlet temp detector would isolate letdown, but the normal letdown path is already isolated.
Plausible; Examinee may confuse the flow path for Additional Purification with Excess Letdown, which would be affected by this failure.

D - WRONG; PIC-201, Letdown Back Pressure Controller, is operating in MANUAL while on additional purification and will not be affected by the pressure transmitter failure.
Plausible; Examinee may recognize that this failure would be correct, if the valves were operating in the normal mode of automatic.

References

ARP 2590B-033

NO Comments or Question Modification History at this time.

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

Number K4.03 **RO** 2.8 **SRO** 2.9 **CFR Link** (CFR: 41.7)

Knowledge of CVCS design feature(s) and/or interlock(s) which provide for the following: Protection of ion exchangers (high letdown temperature will isolate ion exchangers)

Question #: **31** | **Question ID: 2016012** | ☑ **RO** ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The plant has been operating at 100% power for 100 days.

After the crew completion of SP 2654B "Forcing Pressurizer Sprays" the Pressurizer steam space vent line up was not secured.

10 days later Letdown isolates inadvertently.

All immediate Operator actions were completed.

Over the next hour which of the following actions are required to be completed and why?

☐ **A** Ensure RCP Bleedoff aligned to the Equipment Drain Sump Tank (EDST) to prevent over pressurization of the VCT due to the PZR vent line still aligned.

☑ **B** Reduce Turbine load due to lowering RCS Cold leg Temperature per OP 2204 "Load Changes" Attachment 6 "Cold leg Temperature vs. Power Program".

☐ **C** Due to charging pumps being secured, the reactor must be tripped when PZR level lowers below trip criteria IAW AOP 2512 "Loss of All Charging".

☐ **D** Commence a Plant Shutdown within one hour due to the application of Tech. Spec. 3.0.3 for both ECCS subsystems not OPERABLE.

Question Misc. Info: MP2*LOIT NRC-2016

Justification

"A" - WRONG; VCT level will continue to rise and Auto divert at 88% because RCP Bleedoff does not isolate when letdown isolate, there are no requirements to shift RCP Bleedoff to the EDST.
Plausible; The operator may remember from OP 2304A when isolating Charging and Letdown from rated conditions to refer to manual operation of RCP Bleedoff due to VCT Auto diverting to minimize perturbations on the system.

"B" - CORRECT; SP 2654B discussion states with the vent line to the VCT for extended periods boron will concentrate in the PZR as steam and non condensable gases are vented to the VCT, as PZR level lowers with no charging the excess boron coming from the PZR will cause RCS temperature to Lower requiring Operators to adjust Turbine Load to maintain TCOLD.

"C" - WRONG; AOP 2512 does apply, the charging pumps are still able to charge into the RCS therefore a loss of only Letdown then the Reactor Trip Criteria does not apply.
Plausible; The operator knows that the Trip Criteria of AOP 2512 applies because ARP requires the Operator to "Go To" AOP or immediate Actions has the Operator GoTo AOP 2512. One hour of RCP seal leakage at 4 gpm would not reduce PZR level to 55% the required AOP 2512 trip point and if require the Operators would cycle Charging pumps as necessary to maintain PZR level.

"D" - WRONG; AOP 2585 Immediate Operator Actions requires charging pumps be placed in PTL a plant shutdown is not required because any charging pump can be restored by removing the handswitch out of PTL
Plausible; The operator may think that a plant shutdown is required because LCO 3.0.3 does apply when Charging pump handswitches are in PTL but procedurally the handswitches are restored to normal after start to allow a second backup charging pump to cycle on PZR level.

References

SP 2654B

NO Comments or Question Modification History at this time.

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

Number K6.01 **RO** 3.1 **SRO** 3.3 **CFR Link** (CFR: 41.7 / 45.7)

Knowledge of the effect of a loss or malfunction on the following CVCS components: Spray/heater combination in PZR to assure uniform boron concentration

Question #: **32** **Question ID: 2016013** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☑ **Lower Order?**

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

The plant is in Mode 6 during a Refueling Outage with the following conditions:

- Reactor Vessel Head removed
- Refuel Pool level at 36' 6".
- Fuel movement suspended

Under which of the following does Technical Specifications allow suspension of Shutdown Cooling flow for up to one hour in an 8 hour period?

☑ **A** Maintenance performed on valves in the common SDC suction line provided any RCS additions have a Boron Concentration greater than refueling concentration.

☐ **B** Integrated Emergency Safeguard Actuation testing provided any RCS additions have a Boron Concentration greater than refueling concentration.

☐ **C** Shifting of Protected Train facility when Service Water headers are cross-tied provided any RCS additions are less than 44 gallons per minute from any source.

☐ **D** Performing local leak rate testing of SI-709 "SDC Suction Isolation" provided any RCS additions are less than 44 gallons per minute from any source.

Question Misc. Info: MP2*LOIT NRC-2016

Justification

"A" CORRECT**;** Tech. Spec. 3.9.8.1 The required shutdown cooling train may not be in operation for up to 1 hour per 8 hour period provided no operations are permitted that would cause introduction of coolant into the Reactor Coolant System with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1.

"B" WRONG; Integrated Emergency Safeguard Actuation testing is not listed as an reason to suspend SDC flow. Plausible; The Examinee may assume the allowance for securing the LPSI injection pump (SDC) to allow for integrated testing to verify that the LPSI pump gets an automatic start signal.

"C" WRONG**;** The LCO does not allow for the suspension of SDC flow even when the Ultimate heat sink is secured to cross tie Service Water headers during an outage.
Plausible; The examinee may assume that when securing Service water headers that without a heat sink temporarily securing SDC flow will ensure the RBCCW system will not exceed any limits when the ultimate heat sink flow is stopped.

"D" WRONG**;** While Local Leak Rate test is correct, no RCS additions allowed less then refueling boron concentration. Plausible; Although Local Leak Rate test is correct, Examinee may recall Dilution requirements when <300°F to be limited to 1 charging pumps without restriction of the source.

References

TS 3.9.8.1

NO Comments or Question Modification History at this time.

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

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

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

Question #: **33** | ***Question ID:*** ***81728*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ **Student Handout?** | ☑ **Lower Order?**

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

How is power to the SIT outlet isolation valves configured when in Modes 1 and 2?

☐ **A** Disconnect switches between the breaker and the Motor Operator maintain power isolated from the valve but allow control power to the indicating lights.

☐ **B** The breakers are maintained in the open position and the lights are illuminated by control power from vital DC panels.

☐ **C** The breakers are maintained in the closed position to power indicating lights and the valves are locked in MANUAL OPEN.

☑ **D** The breaker closing coils are removed and the breaker is maintained in the ON position to provide power to the indicating lights.

Question Misc. Info: MP2*LOIT SIT, MOV, Control Power, NRC-2016

Justification

A - WRONG; The indication lights for these valves is supplied through the actual MCC breaker contacts.
Plausible; Examinee may believe these valves are set up similar to SI-652 on the same panel, which has an open disconnect switch but an energized light.

B - WRONG; The indication lights for these valves is not supplied by vital AC or DC control power.
Plausible; Examinee may confuse the SIT valves with other valves on C-01 that get their indication from 120/125 control power.

C - WRONG; The breakers are closed to energize the indication, but the valves are NOT locked open in manual.
Plausible; Examinee may confuse the SIT valves with SI-306 on C-01, which is maintained locked open in manual.

D - CORRECT; The SIT MOVs never had a manual disconnect installed. As such, they must be kept from closing with a hot short or other circuit malfunction by removing the closing coil, because opening the MCC breaker would de-energized the indicating lights on C-01.

References

OP 2310

NO Comments or Question Modification History at this time.

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

Number K2.02 **RO** 2.5* **SRO** 2.9 **CFR Link** (CFR: 41.7)

Knowledge of bus power supplies to the following: Valve operators for accumulators

Question #: **34** **Question ID: 2016014** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☐ **Lower Order?**

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

The plant tripped from 100% power due to a Loss Of Coolant Accident (LOCA) and EOP 2532, Loss Of Coolant Accident, has been entered.

The following conditions now exist:

- RCS pressure is 40 psia.
- CETs are currently reading 267 °F
- Both LPSI pumps are off.

Which of the following sets of post accident indications (taken individually) would indicate that the status of the LPSI pumps meets the EOP requirements?

☐ **A** PPC indicates CTMT pressure was 45 psig 4 hours ago, but C-01 indicates CTMT pressure is now 7.7 psig and lowering.

☐ **B** C-08 indicates 24C is de-energized due to a bus fault and C-01 indicates "C" HPSI pump is aligned for hot leg injection.

☑ **C** C-01 indicates two HPSI pumps are injecting at full flow and the RWST level is 9.0% and stable.

☐ **D** PPC indicates that Pressurizer level is stable at 18% and the RVLMS level is now at 80% and rising.

Question Misc. Info: MP2 LOUT, E32-01-C, MB-4732, NRC-2016

Justification

A - Wrong; These are indications that would allow securing of the CTMT Spray pumps, but not the LPSI pumps.
Plausible; Examinee may confuse termination criteria as the stated conditions due meet criteria contained in the applicable EOP.

B - Wrong; These indications would require a LPSI pump be restarted for core heat removal while the HPSI pump is used for hot leg injection.
Plausible; Examinee may confuse two different flow paths used for hot leg injection.

C - CORRECT; Due to the tripping of LPSI Pumps on SRAS at 9.5% RWST level (level is stated as stable, indicating no flow from RWST)

D - Wrong; SI termination/throttling criteria are not met based on RCS subcooling (indications given in the stem).
Plausible; Examinee may focus on indication of RCS inventory recovery, which is one of the criteria for securing LPSI pumps (i.e.; they're not needed).

References

EOP 2532

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.4.3 **RO** 3.7 **SRO** 3.9 **CFR Link** (CFR: 41.6 / 45.4)

Ability to identify post-accident instrumentation.

Question #: **35** | **Question ID:** 8054464 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

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

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

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

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

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

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

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

Question Misc. Info: MP2*LOIT*0634 [002 RCS-01-C 4928] (8/15/96) 2301, RCS, NRC-2008 [K/A; 007, A1.03], NRC-2016

Justification

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

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

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

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

References

OP 2301A

Comments and Question Modification History

Minor wording changes to the four choices to remove repeated words. - RLC

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

Number A1.02 **RO** 2.7 **SRO** 2.9 **CFR Link** (CFR: 41.5/45.5)

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

Question #: **36** | **Question ID:** **90020** | ☑ **RO** ☐ **SRO** | ☐ Student Handout? | ☐ Lower Order?

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

The plant is at 100% power with the following conditions:

- Bus 24E is aligned to Bus 24C.
- "B" RBCCW Pump was just placed in service
- "A" RBCCW Pump in PTL.
- "B" RBCCW Pump SIAS/LNP Block Switch is still in the BLOCK position.

What is the impact of this on plant operations?

☐ **A** No impact on plant operations.
This switch only has an impact when "B" RBCCW Pump is powered from one facility and is supplying the other facility.

☐ **B** "A" Diesel Generator is NOT OPERABLE.
A D/G becomes inoperable any time two RBCCW Pumps are aligned to start on that Facility due to a SIAS or LNP.

☑ **C** "A" RBCCW Header is NOT OPERABLE.
With the pump operating and the SIAS/LNP Block switch in the BLOCK position it will not restart for an emergency.

☐ **D** Facility 1 ECCS safety injection is NOT OPERABLE.
This results in entry into TSAS 3.5.2 requiring a plant shutdown be initiated within one hour.

Question Misc. Info: MP2*LOIT, RBCCW, 2330A

Justification

A - WRONG; The switch is designed to ensure two RBCCW pumps, powered by the same facility (i.e.; same EDG) do not both get a SIAS/LNP start signal. This is a possibility if while swapping RBCCW pumps, a SIAS/LNP occurred during the brief moment that both pumps were running. With the switch in "block", the "B" RBCCW pump auto start signal is still defeated.
Plausible; Examinee may recall the cross facility-alignment mentioned is a suggested action for loss of the running pump in AOP 2564, "Loss of RBCCW", to prevent a otherwise required plant trip on loss of RBCCW.

B - WRONG; The EDG operability is not affected because the "A" RBCCW pump handswitch is in PTL.
Plausible; Examinee may recall the intended purpose of the switch is to ensure EDG operability.

C - CORRECT; With the "B" RBCCW Pump as the Facility 1 Pump, the block switch must be in the NORM position to allow the pump to start on a SIAS or LNP.

D - WRONG; The entire Facility 1 ECCS is not inoperable simply because the RBCCW system is inoperable. This would be an example of "cascading Tech. Specs. Action Statements"
Plausible; Examinee may feel that because the RBCCW system is the required heat sink in CTMT for all design base accidents, calling the system inop would defeat the operability basis for ECCS systems on that facility.

References

OP 2330A

Comments and Question Modification History

"D" distracter change to TSAS 3.5.2 from 3.0.3

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

Number A3.01 **RO** 3.2* **SRO** 3.0 **CFR Link** (CFR: 41.7 / 45.5)

Ability to monitor automatic operation of the CCWS, including: Setpoints on instrument signal levels for normal opera-tions, warnings, and trips that are applicable to the CCWS

Question #: **37** | **Question ID:** 8000067 | ☑ RO ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 0 | ☑ Selected for Exam | **Origin:** Bank | ☐ Past NRC Exam?

The following initial plant conditions exist:
- 100% steady-state.
- Channel "Y" Pressurizer Level and Pressure Control selected as the controlling channels.

Then, VR-21 deenergizes.

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

- ☐ **A** Channel "Y" pressurizer pressure input would fail low, causing pressure control to slowly raise actual pressurizer pressure.
- ☐ **B** Channel "Y" pressurizer level input would fail low, causing pressurizer level control to slowly raise actual pressurizer level.
- ☐ **C** Pressurizer backup heaters would deenergize if on, RCS pressure would stabilized on the proportional heater output.
- ☑ **D** All pressurizer heaters would deenergize, spray valve bypass flow would cause RCS pressure to continue to lower.

Question Misc. Info: MP2*LOIT 2304A, PLPCS, VR-21, 2504B, NRC-2008, NRC-2016

Justification

A - WRONG; PZR pressure input is powered by VA-20 (VIAC), NOT VR-21, a non-vital power supply.
Plausible; Examinee may believe pressure and level control circuits are powered by the same type of power.

B - WRONG; Although the Ch. "Y" PZR level control circuit is normally powered from VR-21, the level transmitter (input) is powered by a VIAC (VA-20).
Plausible; Examinee may assume the PZR level transmitting circuit is powered by the same source as the level control circuit.

C - WRONG; The loss of VR-21 would cause a loss of ALL heaters due to the impact on the low level trip circuit.
Plausible; Examinee may focus on the effect of VR-21 on the backup heaters due to the impact on the high pressure trip circuit, which trips all Backup Heaters (only) and prevents them from being energized.

D - Correct; The Pressurizer Heater Selector switch is normally in the "Both" position, which means a loss of VR-11 OR VR-21 will cause all PZR heaters to deenergize due to the failure of the heater low level cutout circuit. The recovery of the heaters requires the operators to de-select the failed/de-energized circuit (select Ch. "X" only) and reclose both Proportional heater breakers.

References

AOP 2504B

Comments and Question Modification History

Modified choice "C" slightly to separate it further from other evaluated concepts. - RLC

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

Number K3.01 **RO** 3.8 **SRO** 3.9 **CFR Link** (CFR: 41.7 / 45.6)

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

Question #: **38** | **Question ID:** **2016015** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

A plant startup is in progress with power presently at 20% and pressurizer sprays are being forced. Then, the Spray Valve Controller, HIC-100E2, on C-21 fails to "manual" mode with a minimum output.

Which of the following states the effect of this controller failure and the action required to maintain pressurizer pressure at the desired value?

☐ **A** Spray Valve controller, HIC-100E, on C-03 will shift to MAN and Spray Valve RC-100E will fails "as-is", partially open.
The in-service pressure controller must be shifted to manual and its ouput lowered as needed to control pressure.

☐ **B** Spray Valve controller, HIC-100E, on C-03 will shift to MAN and Spray Valve RC-100E will fails "as-is", partially open.
The auto setpoint on the in-service pressure controller must be lowered as needed to control pressure.

☐ **C** Spray Valve controller, HIC-100E, on C-03 will remain in AUTO and Spray Valve RC-100E will go full closed.
The in-service pressure controller must be shifted to manual and its ouput lowered as needed to control pressure.

☑ **D** Spray Valve controller, HIC-100E, on C-03 will remain in AUTO and Spray Valve RC-100E will go full closed.
The auto setpoint on the in-service pressure controller must be lowered as needed to control pressure.

Question Misc. Info: MP2*LOIT, RCS, C-21, MAIN SPRAY, PPLCS, IH-2011
, {ILT 2011 Audit Q#2/Access#=1100102/NITIMS#=88824}

Justification

A - WRONG; HIC-100E on C-03 is "upstream" of HIC-100E2 in the control circuit so that taking control of the valve at C-21 will override the controller output on C-03. However, with this type of alignment, HIC-100E is blind to any change in HIC-100E2.
Plausible; Examinee may believe the controllers on the two panels function in tandem like the Main Feedwater Reg. Valve Controllers, where the master controller sees the end result of both main and bypass valve controller outputs.

B - WRONG; The spray valves normally "float" partially open during the evolution of forcing spray flow. However, this failure will not cause the valve to "lock-in" to the specific position it was at when the circuit failure occurred. It will go to the position demanded by the failed output signal of HIC-100E2.
Plausible; Examinee may believe the valve control circuit is similar to other Foxboro IA controllers in that a failure to manual can lock in the signal to its last good value, or the MFRV circuit that would cause the main feed reg valve to "lock-up" as-is on a major control circuit failure, therefore the pressure control setpoint must be lowered to allow the functioning spray valve to open.

C - WRONG; Although the pressure controller is strictly a proportional controller, simply adjusting the auto setpoint lower will raise the signal going to the functioning spray valve and return PZR pressure to the desired value.
Plausible; Examinee may believe that because the failure affects a cascaded controller in the circuit, manual control is necessary to ensure proper pressure control and the output must be lowered to lower pressure.

D - CORRECT; The C-21 controller is downstream of the C-03 controller; therefore, the spray valve will go FULL closed and the C-03 controller will NOT change. With the one of the two spray valves going full closed (as opposed to floating partially open), PZR pressure will rise until the operating spray valve opens enough to account for the manually energized heaters. As the pressure controller is strictly a proportional controller, the auto setpoint must be lowered to return PZR pressure to the desired value.

References

PLC-01-C

Comments and Question Modification History

Revised based on initial validation feedback. - RLC

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

Number A2.02 **RO** 3.9 **SRO** 3.9 **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Spray valve failures

Question #: **39** | **Question ID:** **1654407** | ☑ **RO** ☐ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**
Rev. **0** | ☑ **Selected for Exam** | **Origin:** **Bank** | ☐ **Past NRC Exam?**

A middle-of-life (MOL) reactor startup is in progress, with all plant systems and components functioning as designed for this condition.

As power is raised above the point of adding heat, an inadvertent positive reactivity addition results in a sudden, ten (10) dpm Start Up Rate (SUR).

Which one of the following instruments will sense this transient and provide DIRECT input to an automatic function that will ensure the specified limit is not exceeded?

☐ **A** Pressurizer pressure detectors, to prevent exceeding RCS high pressure limit.

☐ **B** RCS temperature detectors, to prevent exceeding DNBR or fuel centerline temperature limit.

☐ **C** Control Channel Nuclear Instruments, to prevent exceeding clad surface temperature limit.

☑ **D** Linear Power Nuclear Instruments, to prevent exceeding TM/LP or high pressure limit.

Question Misc. Info: MP2 LORT LOIT RPS, NRC-2016

Justification

A - WRONG; PZR sprays are being forced, which would cause the PZR Pressure Control System to immediately respond to any rise in RCS pressure, delaying a trip on RCS high pressure. Therefore, at the given SUR, temperature input from power will not raise pressure fast enough before power exceeds any Safety Limits.
Plausible; Examinee may focus on the temperature input from the power increase, and may think that the actual power rise will be buffered by feedback from the Power Defect at MOL, leaving RCS pressure as the tripping parameter.

B - WRONG; During a reactor startup, The Main Steam System is in operation with condenser dump valves set to control RCS temperature in automatic. The steam dumps would respond very quickly and buffer any rise in RCS temperature, delaying any input to the TM/LP calculation. Therefore, RCS temperature will not rise fast enough with the high SUR.
Plausible; Examinee may remember the Tech. Spec. calculation for TM/LP (DNBR protection) setpoint multiplies each degree of RCS temperature rise by 14.28, which would cause the trip setpoint to quickly gain on the actual pressure and cause a trip.

C - WRONG; Control channel NIs can only assist by actuating the Diverse Scram System (DSS) and cause an early initiation of Aux. Feedwater Actuation if two safety channels of RCS pressure are exceeding their trip setpoints. This is designed to mitigate an ATWS combined with a loss of all normal feedwater (loss of heat sink).
Plausible; Examinee may focus on the DSS as it is designed to mitigate an unusual power excursion that is not mitigated by RPS.

D - CORRECT; Technical Specification Bases 2.2.1 states, "The Power Level-High Trip provides reactor core protection against reactivity excursions which are too rapid to be protected by a Pressure-High or Thermal Margin/Low Pressure trip.

References

TS 2.2.1

NO Comments or Question Modification History at this time.

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

Number K4.02 **RO** 3.9 **SRO** 4.3 **CFR Link** (CFR: 41.7)

Knowledge of RPS design feature(s) and/or interlock(s) which provide for the following: Automatic reactor trip when RPS setpoints are exceeded for each RPS function; basis for each

Question #: **40** | **Question ID: 4003900** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The plant is in Mode 5 and refueling has just been completed.

Containment Purge is in operation.

Then, VA-30 is lost when the panel's main breaker fails open.

What is the effect on the ESAS system?

☐ **A** Sensor Cabinet "C" is de-energized.
Many sensor modules (SIAS, MSI, CSAS etc.) have triggered on Sensor Cabinet "C".
CPVIS is actuated on ESAS Actuation Cabinet "5" only.

☐ **B** Sensor Cabinet "C" is energized.
There are NO sensor modules (SIAS, MSI, CSAS etc.) triggered on any sensor cabinet.
CPVIS is not actuated on either ESAS Actuation Cabinet.

☐ **C** Sensor Cabinet "C" is de-energized.
There are NO sensor modules (SIAS, MSI, CSAS etc.) triggered on any sensor cabinet.
CPVIS is actuated on both ESAS Actuation Cabinets "5" and "6".

☑ **D** Sensor Cabinet "C" is energized.
Many sensor modules (SIAS, MSI, CSAS etc.) have triggered on Sensor Cabinet "C".
CPVIS is actuated on both ESAS Actuation Cabinets "5" and "6".

Question Misc. Info: MP2*LOIT, ESAS, MB-02467, MB-2470, MB-02469, MB-03117, NRC-2016

Justification

The effects of a loss of VA-30 on ESAS requires the student to understand both ESAS and the Spec. 200 cabinets. All ESAS sensor cabinets are dual powered, therefore a loss of only VA-30 does not cause any ESAS sensor cabinet to deenergize. The actuation cabinets are powered from VA-10 and VA-20 so they remain powered during a loss of VA-30. The result is that ESAS is fully operational. The problem is in the Spec 200 cabinets, which are not dual powered. A loss of VA-30 causes the ESAS detectors on channel "C", which get there power through Spec. 200, to become deenergized. When ESAS sensors deenergize, they send a trip signal to their sensor cabinet and trigger the applicable sensor module. Because only 1 channel is tripped, this does not meet the 2 of 4 logic needed to actuate SIAS, MSI CSAS etc. However, the one exception is CPVIS which only requires a 1 out of 4 logic to actuate.

A - WRONG; Actuation of CPVIS requires a 1/4 logic, and once met, will trigger a full ESAS actuation (both facilities).
Plausible; Examinee may recognize that CPVIS will actuate, but assume it is facility dependent based on the abnormal logic required (not 2/4 like all other actuations).

B - WRONG; Loss of VA-30 causes the detectors for safety channel 'C' to deenergize due to Spec 200, resulting in several sensor modules being triggered on ESAS Sensor Cabinet 'C'.
Plausible; Examinee may focus on the "backup" power supply to the ESAS Sensor Cabinets and assume this would prevent any sensor modules from triggering.

C - WRONG; Sensor Cabinet "C" is not de-energized because it is dual powered (normal: VA-30, backup: VA-20).
Plausible; Examinee may understand the effect of a loss of power to safety channel "C" based on the loss of power to the Spec 200 cabinets, but miss the effect of the backup power supply to the ESAS Sensor Cabinets.

D - CORRECT; Cabinet "C" is dual powered and therefore energized. Sensor modules on channel 'C' are triggered because of SPEC 200. CPVIS is actuated on both facilities because ESAS Sensor Cabinet 'C' has an actual triggered signal and it meets the 1 of 4 logic for CPVIS actuation.

References

AOP 2504E, ARP 2590A-143, OP 2384

NO Comments or Question Modification History at this time.

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

Number K2.01 **RO** 3.6* **SRO** 3.8 **CFR Link** (CFR: 41.7)

Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control

Question #: **41** | **Question ID: 2016028** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The plant is operating in Mode 1 when the following Alarm is received:

- “CTMT AIR RECIRC FAN “A” VIBRATION HI” C01 A-4

The Reactor Operator refers to the Alarm Response Procedure (ARP) and Presses “CAR FAN VIB RESET A & C" button (C-01) which did not reset the Alarm.

2 minutes later the “A” CTMT AIR RECIRC FAN trips.

Which of the statements correctly describes the Operator’s actions in accordance with the applicable ARP?

☐ **A** Start an idle CTMT AIR RECIRC FAN in Fast Speed and verify “A” CTMT Spray Pump OPERABLE to meet CTMT Cooling train LCO.

☐ **B** Ensure highest CTMT air temperature is below the Tech. Spec limit, and verify “A” CTMT Spray Pump OPERABLE to meet CTMT Cooling train LCO.

☐ **C** Ensure highest CTMT air temperature is below the Tech. Spec limit, and enter the Tech. Spec. action for 1 CTMT Cooling train NOT OPERABLE.

☑ **D** Start an idle CTMT AIR RECIRC FAN in Fast Speed and enter the Tech. Spec. action for 1 CTMT Cooling train NOT OPERABLE.

Question Misc. Info: MP2*LOIT CCS-01-C, ARP-2590A-013, NRC-2016

Justification

A - WRONG; Requires entry into the Tech. Spec. action for 1 CTMT Cooling train NOT OPERABLE.
Plausible; The examinee may understand that a CTMT Spray pump meets design criteria however does not meet the wording of the applicable LCO when the fan is not OPERABLE.

B - WRONG; The Tech. Spec. for Containment Temperature is for the AVERAGE temperature and not the Highest. Requires entry into the Tech. Spec. action for 1 CTMT Cooling train NOT OPERABLE which consist of 2 CAR fans and 1 Containment Spray pump. When the CAR fan trips it is considered not OPERABLE.
Plausible; The examinee may understand that a CTMT Spray pump meets design criteria however does not meet the wording of the applicable LCO when the fan is not OPERABLE.

C - WRONG; The Tech. Spec. for Containment Temperature is for the AVERAGE temperature and not the Highest. ARP directs starting an idle fan, normal operations requires 3 fans in fast.
Plausible; The examinee may not know the requirements for 3 fans in operation because only 1 Facility (2 CAR Fans) is required to meet accident analysis and may consider that maintaining temperature in Containment ensure continued operation.

D - CORRECT; ARP directs starting an idle fan with normal operations requiring 3 fans in fast speed with the additional breaker trip the Fan is not considered OPERABLE therefore the crew must log into the applicable LCO for cooling train not OPERABLE.

References

ARP 2590A-013, ARP 2590A-009, TS 3.6.2.1

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.4.50 **RO** 4.2 **SRO** 4.0 **CFR Link** (CFR: 41.10 / 43.5 / 45.3)

Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Question #: **42** | **Question ID:** 1100020 | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

A Large Break LOCA has occurred from 100% power operation concurrent with a loss of Bus 24C. SIAS, CIAS, EBFAS, MSI, and CSAS have all automatically actuated.

- "B" Containment Spray header flow indicates 1,210 gpm.
- RBCCW flow to each operating CAR Cooler is 2,100 gpm.

What is the status of the Containment Cooling System with regard to its ability to perform its intended function?

☐ **A** The "B" Containment Spray header has more than the required design flow. With two CAR Coolers in service, cooling is sufficient to ensure Containment temperature and pressure will remain within design limits.

☐ **B** The Containment Spray System does NOT have adequate flow to establish an effective spray pattern; therefore, the Iodine concentration in the Containment atmosphere will remain high until adequate flow is established.

☐ **C** The Containment Spray System and CAR Coolers are presently providing adequate Containment cooling; however, when SRAS occurs, Containment Spray flow will NOT be adequate to maintain core cooling.

☑ **D** The "B" Containment Spray header has less than the required design flow. With only two CAR Coolers in service, cooling is NOT sufficient to ensure Containment temperature and pressure will remain within design limits.

Question Misc. Info: MP2*LOIT, CS, CTMT Spray, 2532, 2309, NRC-2011 [026, K3.01], NRC-2016

Justification

A - WRONG; The "B" Containment Spray header has less than the design (procedural) limit of 1300 gpm. With Bus 24C deenergized, only two CAR Coolers are available. This combination of CAR Coolers and Containment Spray with less than the design flow rate does NOT guarantee that Containment temperature and pressure limits will be maintained less than design limits.
Plausible: If the examinee does NOT know the minimum Containment Spray flow limit, then one Containment Spray header and two CAR Coolers are adequate to ensure Containment temperature and pressure will remain within design limits.

B - WRONG; With a lower than minimum flow, the spray pattern is likely affected; however, Iodine scrubbing of the Containment atmosphere is NOT the overriding function of the Containment Cooling System. Lower than design flow will impact the ability of the Containment Cooling System to ensure Containment temperature and pressure remain below design limits.
Plausible: Iodine scrubbing is a function of the Containment Spray System. The examinee may feel that two CAR Coolers is adequate to provide the required Containment Cooling and that Containment Spray is necessary to reduce Containment atmosphere Iodine concentration, limiting the radioactive release to the environment.

C - WRONG; The Containment Cooling System is NOT providing adequate heat removal from Containment due to low flow in the "B" Containment Spray header, the loss of "A' Containment Spray, and the loss of two CAR Coolers.
Plausible: If Containment Spray does NOT meet the termination criteria when SRAS initiates, then core cooling may be negatively impacted. If the examinee does NOT know the minimum Containment Spray flow limit, then one Containment Spray header and two CAR Coolers are adequate to ensure Containment temperature and pressure will remain within the design limits.

D - CORRECT; The minimum design Containment Spray flow is 1300 gpm. The design of the Containment Cooling System is such that two fully functioning CAR Coolers and one fully functioning Containment Spray System are necessary to prevent exceeding design Containment temperature and pressure limits.

References

EOP 2525

NO Comments or Question Modification History at this time.

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

Number A1.06 **RO** 2.7 **SRO** 3.0 **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 CSS controls including: Containment spray pump cooling

Question #: **43** | **Question ID:** **53438** | ☑ **RO** ☐ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

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

During a plant startup, the main feedwater pump turbines are initially driven by main steam, but eventually are driven by extraction steam ("cross-over" steam).

Choose the statement which correctly describes how the changeover in driving steam occurs.

☐ **A** Remote switch manipulations performed by operators at C05 S/G Feed Pump insert.

☑ **B** Automatically occurs dependent upon SGFP turbine load and available steam pressure.

☐ **C** Automatically accomplished by pressure detectors which actuate the turbine control valves.

☐ **D** Remote valve manipulation performed by operators at C06/7 opening extraction steam valve.

Question Misc. Info: MP2*LORT*3119 [059 MFW-01-C 4251] (1/14/97) 2321, MFW, SGFP, NRC-2016

Justification

A - WRONG; Cross over steam supply to the MFP turbine is a function of the throttle valve positions as the governor requires more steam the cross over throttle valve open first but, does not have enough pressure at low power therefore the throttles open further until the main steam throttles open.
Plausible; Both steam inlet valves are displayed on C05 the operator may believe that the MFP insert controls both sets of valves from C05.

B - CORRECT; Cross over steam supply to the MFP turbine is a function of the throttle valve positions as the governor requires more steam the cross over throttle valve open first but, does not have enough pressure at low power therefore the throttles open further until the main steam throttles open.

C - WRONG; Pressure sensor for cross over steam is an input to High and Low Load valves for MSRs
Plausible; The Examinee may confuse the pressure detector function with controlling MFP turbine steam valves.

D - WRONG; Cross over steam supply to the MFP turbine is a function of the throttle valve positions.
Plausible; The examinee may think that the extraction steam isolation valves located on C06/7 controls the steam supply to the MFP turbine.

References

OP 2321

Comments and Question Modification History

"A" modified to add SGFP insert and "D" N/P modified from local valve operations to remote extraction steam valve manipulations from C06/7. djj

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

Number K1.08 **RO** 2.7* **SRO** 2.9* **CFR Link** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Knowledge of the physical connections and/or cause-effect relationships between the MRSS and the following systems: MFW

Question #: **44** | **Question ID: 2016038** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

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

A plant start-up is in progress after a forced outage.

Which of the following describes when the Main Feedwater Regulating Valve Bypass Valves are required to be closed, and the bases for the requirement?

☑ **A** Prior to exceeding 25% power, to ensure the analysis for an Excess Steam Demand Event inside Containment is not challenged.

☐ **B** Prior to exceeding 25% power, to ensure the analysis for Shutdown Margin on an Excess Steam Demand Event is not challenged.

☐ **C** Prior to exceeding 25% open on the Main Feedwater Regulating Valve, to ensure the analysis for an Excess Steam Demand Event inside Containment is not challenged.

☐ **D** Prior to exceeding 25% open on the Main Feedwater Regulating Valve, to ensure the analysis for Shutdown Margin on an Excess Steam Demand Event is not challenged.

Question Misc. Info: MP2*LOIT* 2204, NRC-2016

Justification

A - CORRECT; OP 2204, Precaution. 3.6 To remain within the main steam line break inside Containment analysis, opening FRV bypass valve(s) is not allowed when greater than 25% power. The FSAR reference states that the analysis only took into account a MFRV Bypass being opened at or below 25% power

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

C - WRONG; The MFRV would automatically open as power is raised, based on the Bypass Valve's inability to maintain S/G level. However, the Bypass Valve must be closed before exceeding 25% power, not 25% valve position, which would be a much higher power level.
Plausible; Examinee may recall the 25% number but assume it was based on valve position, which directly correlates to the amount of feed flow to the effected S/G.

D - WRONG; Bypass Valve must be closed prior to exceeding 25% power, due to the affect on CTMT pressure from the extra feed water flow.
Plausible; Examinee may recall the 25% number but assume it was based on valve position and the affect the extra feed water would have on SDM.

References

OP 2204

Comments and Question Modification History

Underlined the words power and open.djj

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

Number A1.03 **RO** 2.7* **SRO** 2.9* **CFR Link** (CFR: 41.5/45.5)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW controls including: Power level restrictions for operation of MFW pumps and valves.

Question #: **45** **Question ID:** **2016041** ☑ **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 an Excess Steam Demand Event.

On the trip, a circuit malfunction triggered an inadvertant Auxiliary Feedwater Actuation (AFAS with no time delay).

During the performance of EOP 2525, Standard Post Trip Actions, under which of the following safety functions is guidance given to mitigate the inadvertant Auxiliary Feedwater actuation?

☐ **A** RCS Pressure Control

☐ **B** Core Reactivity Control

☐ **C** RCS Inventory Control

☑ **D** RCS Heat Removal

Question Misc. Info: MP2*LOIT 2536, 2525, AFAS, AFW, MSLB, ESD, NRC-2016

Justification

A - WRONG; Pressure control will be affected but the actions to mitigate given here address the PZR pressure control system.
Plausible; Examinee may focus on the loss of RCS pressure, which will result in SI flow into an already full RCS.

B - WRONG; Reactivity control will be affected but the actions to mitigate given here address boric acid injection.
Plausible; Examinee may focus on an important concern with an ESD, specifically reactor restart.

C - WRONG; Inventory control will be affected but the actions to mitigate given here address Safety Injection flow.
Plausible; Examinee may focus on the excess amount of inventory that may result due to SI flow and the RCS pressure drop.

D - CORRECT; The excess feed flow to the affected S/G will cause a more severe RCS cooldown (reactivity) and shrinkage (inventory control), which will inturn impact PZR pressure (pressure control). However, EOP-2525 addresses the excess feed under RCS Heat Removal Safety Function when the operators evaluate the abnormal drop in RCS temperature post-trip.

References

EOP 2525

NO Comments or Question Modification History at this time.

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

Number K3.01 **RO** 4.4 **SRO** 4.6 **CFR Link** (CFR: 41.7 / 45.6)

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

Question #: **46** | **Question ID:** **53551** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

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

A plant startup from a mid-cycle maintenance outage is in progress and the reactor has just been declared critical.

Shortly after going critical, the SPO inadvertently overfeeds the S/Gs, and then regains control.

This action results in average RCS temperature lowering to 519 °F and then stabilizing.

Which of the following operator actions should be taken, based on administrative requirements?

☑ **A** Determine RCS Tave is within its limit once per hour until Tave reaches or exceeds 525 °F.

☐ **B** Restore Tavg to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

☐ **C** Emergency Boration must be commenced immediately.

☐ **D** Immediately trip the Reactor and carry out EOP 2525.

Question Misc. Info: MP2*LOUT S/U, OP, 2203, TS, NRC-2009, NRC-2016

Justification

A - CORRECT; Per T.S. 3.1.1.5, when ever the reactor is critical, RCS Tavg must be >/= 515°F at all times. However, the T.S. Surveillance Requirements state that if Tavg drops below 525°F, then Tavg must be verified to be >/= 515°F (within its limit) at least once per hour.

B - WRONG; This is the required action if Tavg dropped below 515°F.
Plausible; Examinee may believe the limit for action is < 525°F as that is the reactor startup procedural limit (OP-2202).

C - WRONG; This is the required action if Tavg dropped below 500°F.
Plausible; Examinee may believe an uncontrolled cooldown below the procedural required limit necessitates immediate action to ensure reactivity control.

D - WRONG; This is the required action if the cooldown was not stopped (temperature control is not regained) before Tavg drops below 500°F.
Plausible; Examinee may believe an uncontrolled cooldown below the T.S. limit requires immediate action to ensure plant control.

References

OP 2202, TS 3.1.1.5

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 2.2.38 **RO** 3.6 **SRO** 4.5 **CFR Link** (CFR: 41.7 / 41.10 / 43.1 / 45.13)

Knowledge of conditions and limitations in the facility license.

Question #: **47** | **Question ID: 1655196** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

Millstone Unit 2 is shut down and on the RSST in Mode 3.

A fault in the Millstone 345KV Switchyard causes the loss of the "North" Bus.

What effect would this have on Unit 2 and any actions required to mitigate this event if any?

☐ **A** No effect, continue to monitor RCS parameters IAW OP 2272C “Plant Operation in MODE 3”.

☑ **B** Unit 2 experienced a LNP, transition to EOP 2528 “Loss of Force Circulation” establish NC flow.

☐ **C** Unit 2 Main Transformers lost power to the cooling units, refer to ARP to swap power.

☐ **D** No effect, ensure 15G-7T Unit 2 North Ring Bus tie breaker is open, refer to OP 2351 "345KV".

Question Misc. Info: MP2*LOIT*2523 [062 345-01-C 986] (8/27/96) 2347, SWYD, APP

Justification

"A" WRONG; With a loss of the North Bus Unit 2 loses the RSST causing a LNP, therefore restoring only the Vital Buses from the EDG.
Plausible; The Examinee may think that the RSST is powered from the South Bus preserving Off-Site power allowing the Unit to stay in the current EOP 2526 and monitoring parameters.
"B" CORRECT; The RSST is powered from the 345KV North Bus thus de-energizing the RSST losing the RCPs requiring the crew to transition to EOP 2528 "Loss of Off-Site Power / Loss of Forced Circulation" and ensuring Natural Circulation is established is required knowledge for a RO of knowing entry conditions.
"C" WRONG; Although the Main Tranformer lost power due to the loss of the RSST and all its cooling fans de-energized swapping power supplies would not work because the alternate power supply is also de-emergized.
Plausible; C06/7 AA-50 will alarm on the loss of normal to the Main Transformer causing the loss of cooling fans requiring a manual swap of the power supply to the cooling units which will not power the fans because they are also de-energized.
"D" WRONG; Although it is true that the 15G-7T will open on a loss of the North Bus the requirement for the Crew to verify the breaker opened is only if the breaker failed to trip.
Plausible; The examinee may remember that 15G-7T will trip to isolate the North Bus as part of the tripping scheme for protecting the 4 Off-Site lines.

References

OP 2351, AOP 2502A

NO Comments or Question Modification History at this time.

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

Number A2.04 **RO** 3.4* **SRO** 3.1 **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effect on plant of de-energizing a bus

Question #: **48** **Question ID:** **8000061** ☑ **RO** ☐ **SRO** ☐ Student Handout? ☐ Lower Order?

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

The plant has just started up from a refueling outage and is stable at 30% power on a secondary chemistry hold.
VR-21 is presently aligned to B62 due to ongoing testing of the UPS static switch.

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 Steam Generator and C08 alarm indicating VR-21 is de-energized.

☐ **B** C02/3 alarms indicating RCS Thot and Tcold are abnormally low and both Boric Acid Pumps are de-energized.

☑ **C** C04 alarms indicating Facility 1 Auxilary 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.

Question Misc. Info: MP2*LOIT ESD, CTMT, 120VAC/125VDC, NRC-2008, NRC2016

Justification

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).
Plausible; Examinee may recognize that losing VR-21 affects ADV operation, but confuse the specifics of the effect it will have.

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.
Plausible; Examinee may recognize the need for boric acid injection and that a loss of the BA pumps and the running charging pump would hamper that evolution.

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 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.

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.
Plausible; Examinee may focus on the fact that CS is more effective than CAR fans in mitigating an ESD and believe efforts need to be made to energize the second CS pump or additional CAR fans.

References

OP 2260, AOP 2505B

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 063 D.C. Electrical Distribution

Number K3.02 **RO** 3.5 **SRO** 3.7 **CFR Link** (CFR: 41.7 / 45.6)

Knowledge of the effect that a loss or malfunction of the DC electrical system will have on the following: Components using DC control power

Question #: **49** | **Question ID:** **6055354** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

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

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

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

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

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

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

☐ **B** Place the "AUTO/MAN" switch inside INV-1 to the "AUTO" position to immediately recover VA-10.

☐ **C** Place the "SYNC" switch on #1 static switch to the "OFF" position to immediately recover VA-10.

☑ **D** When INV-5 deenergized, VA-10 deenergized and cannot be recovered under existing conditions.

Question Misc. Info: MP2*LOIT LVD-01-C, 2345, NRC-2016

Justification

A - WRONG: The transfer is blocked from happening until both Normal and Alternate power sources are in synch (energized).
Plausible; Examinee may think that due to the power seeking circuitry the static switch will auto transfer to inverter 1, which is effectively ready to load.

B - WRONG: This action would place #1 static switch in a "normal seeking" mode, but it would not transfer to INV-1 because INV-1 and INV-5 are not in synch.
Plausible; Examinee may think that the static switch is sofisticated enough to sense that the Alternate source is deenergized and ignore the synch fail check.

C - WRONG: Placing the "SYNC" switch to OFF is one of the actions that triggered the actual event that occurred at MP2.
Plausible; Examinee may recognize that the AUTO Sync circuit is preventing the static switch from transferring and believe that turning it off would negate the transfer block.

D - CORRECT: The nomenclature of the "AUTO/MAN" switch inside INV-1 is similar to the "AUTO/MAN" switch OUTSIDE the INV-1 cabinet on the Static Switch. However, their functions are quite different. With the INV-1 switch specified in the stem in the applicable position, the Static Switch will NOT transfer to the Normal power supply and VA-10 will be deenergized. Once it is deenergized, it cannot be re-energized, by procedure, until both the Normal (INV-1) and Alternate (INV-5) power supplies are restored, allowing the synch check circuit to transfer VA-10 to INV-1.

References

AOP 2504C, LVD-00-C

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 063 D.C. Electrical Distribution

Number K4.01 **RO** 2.7 **SRO** 3.0* **CFR Link** (CFR: 41.7)

Knowledge of DC electrical system design feature(s) and/ or interlock(s) which provide for the following: Manual/automatic transfers of control

Question #: **50** | **Question ID: 2016033** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The 'A' Emergency Diesel Generator (EDG) was being run for a surveillance test when it automatically shut down due to a "Crankcase Pressure High" alarm.

The PEO sent to investigate the alarm reports pressing the "alarm acknowledge button" on the local panel (C-38).

Under the existing conditions, if a Loss of Offsite Power were to occur, which of the following describes the status of the 'A' EDG?

☐ **A** The EDG will start, come up to speed and automatically pickup bus loads.

☐ **B** The EDG will start, but will not load until reset locally by the PEO.

☑ **C** The EDG will remain shutdown and not respond to any start signal.

☐ **D** The EDG will remain shutdown and only respond if a SIAS is also triggered.

Question Misc. Info: MP2*LOIT EDG, Trips, Interlocks, NRC-2016

Justification

A - WRONG; The SDR is still armed and would not allow the EDG to start for any reason except loss of DC control power.
Plausible; Examinee may assume the emergency start signal overrides the high crankcase trip in all cases.

B - WRONG; The SDR is still armed and would not allow the EDG to start, which is one of the required actions for the breaker to close.
Plausible; Examinee may believe the interlock only blocks the breaker closure, which has logic requirements in addition to EDG at speed.

C - CORRECT; The high crankcase pressure alarm also triggers an EDG "non-emergency" trip, which will activate the EDG Shutdown Relay (SDR) if the EDG is not running due to an emergency start (LOOP, LNP, or SIAS). The SDR must be reset locally, at the engine skid, after all trip alarms have been reset. As the PEO attempted to reset the SDR before the cause of the trip cleared, it would not have reset and still be armed. In this condition, even an emergency start signal cannot start the EDG, even though the start signal would ordinarily override this non-emergency trip signal.

D - WRONG; Although a SIAS is a separate start signal from the vital power loss signal, it is not capable of overriding the active SDR.
Plausible; Examinee may believe the SIAS EDG start is more encompassing than the power loss start because it is a separate emergency start of the diesels and triggers additional logic and controls not triggered by a power loss signal.

References

SP 2613G, ARP 2591A-005, ARP 2591A

Comments and Question Modification History

Modified stem to reduce 2nd paragraph wording.djj

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

Number A1.04 **RO** 2.8 **SRO** 2.9 **CFR Link** (CFR: 41.5 / 45.5)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ED/G system controls including: Crankcase temperature and pressure

Question #: **51** | **Question ID:** 1000010 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☑ Lower Order?

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

Which of the following contains ONLY those valves which will receive a close signal due to the Steam Generator Blowdown, (SGBD) RM-4262 exceeding its alarm level setpoint?

☐ **A** SGBD Quench Tank pump discharge valve 2-MS-135, SGBD Blowdown Tank Discharge valve 2-MS-15, and the SG sample isolation valves 2-MS-191A & B.

☑ **B** SGBD Quench Tank pump discharge valve 2-MS-135, SGBD Blowdown Tank Discharge valve 2-MS-15, and the SGBD isolation valves 2-MS-220A & B.

☐ **C** SGBD isolation valves 2-MS-220A & B, blowdown sample discharge to secondary sample sink HV-4287/8 SG , and the DIS TO QUENCH TK MS-117A/B.

☐ **D** SG blowdown sample discharge to secondary sample sink HV-4287/8, Discharge to blowdown tank MS-145A/B, and the SGBD Quench Tank pump discharge valve 2-MS-135.

Question Misc. Info: MP2*LOIT SGBD RM, AOP 2569, NRC-2016

Justification

"A" WRONG; MS-191A&B do not receive an isolation signal on a Rad monitor alarm.
Plausible; Examinee may confuse the signal that closes the SG sample isolation valves 2-MS-191A & B which is a CIAS.

"B" CORRECT; SGBD RM high closes all potential discharge paths for elevated activity, not closing the sample isolations allows continued monitoring of SGBD.

"C" WRONG; DIS TO QUENCH TK MS-117A/B does not receive an isolation signal.
Plausible; Examinee may think the Quench Tank gets isolated to prevent an unmonitored release.

"D" WRONG; Discharge to blowdown tank MS-145A/B does not receive an isolation signal.

Plausible; Examinee may think the Blowdown Tank gets isolated to prevent an unmonitored release..

References

ARP 2590H-005

Comments and Question Modification History

Replaced Q#55932 due to conflicks with other questions.

NRC K/A System/E/A **System** 073 Process Radiation Monitoring

Number K4.01 **RO** 4.0 **SRO** 4.3 **CFR Link** (CFR: 41.7)

Knowledge of design feature(s) and/or interlocks which provide for the following: Release termination when radiation exceeds setpoint

RO and SRO Exam Questions (No "Parents" Or "Originals")

Question #: **52** | **Question ID:** 53448 | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

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

Which of the following conditions, by itself, will result in the Control Room Ventilation System shifting automatically into the Recirculation Mode of operation?

☐ **A** Control Room area radiation monitor (RM-7899) reading greater than 10 mr/hr.

☐ **B** Control Room gaseous process radiation monitor (RM-8011) reading greater than 10 mr/hr.

☑ **C** Control Room ventilation duct radiation monitor (RM-9799A) reading greater than 100 mr/hr.

☐ **D** Spent Fuel Pool area radiation monitor (RM-8142) reading greater than 100 mr/hr.

Question Misc. Info: MP2*LORT*3134 [088 BPV-01-C 2760] (11/25/97) 2315A, CRAC, RM, NRC-2016

Justification

Meets the design limits for control room habitability for the operators during accident conditions. This is the only Rad monitor that the setpoint is related to design limits and not radiation release limits.

"A" WRONG; Although the Rad monitor is located inside the Control Room envelope it has no automatic function.
Plausible; The rad monitor located next to C08 alarms locally may cause the examinee to wrongly deduce an automatic function of shifting CRAC to recirculation mode.

"B" WRONG; Although the Rad monitor samples inside the Control Room envelope it has no automatic function.
Plausible; The process rad monitor samples the Control Room envelope may cause the examinee to wrongly deduce an automatic function of shifting CRAC to recirculation mode.

"C" CORRECT; RM-9799A/B, SIAS/CIAS/EBFAS signal and AEAS signal are the only inputs to automatically place CRAC in recirc mode.

"D" WRONG; Although the Rad monitor is part of the AEAS signal to CRAC it requires 2 of 4 area rad monitors to initiate AEAS therefore CRAC to recirc mode.
Plausible; The area rad monitor is part of the AEAS signal to CRAC

References

ARP 2590A-159

Comments and Question Modification History

Placed commas before and after "by itself"

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

Number A1.01 **RO** 3.2 **SRO** 3.5 **CFR Link** (CFR: 41.5 / 45.7)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM system controls including: Radiation levels

Question #: **53** | ***Question ID:*** ***2016022*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The Plant is currently performing a cooldown for a Mid Cycle Shutdown with Service Water injection temperature at 50°F.

Current conditions:

- "A" and "B" RCPs operating
- "A" Shutdown Cooling pump running
- Both SDC heat exchangers in service.
- 2-SI-306 "SDC Total Flow Control Valve" is 50% open
- 2-SI-657 "SDC System HX Flow CNTL" is 50% open
- TCOLD is at 220°F and Stable

The Control Room has suspended the Cooldown while racking down 2 RCP Breakers.

An Operator mistakenly removes Fuse Block for 2-SW-8.1A "A" "RBCCW HX Service Water Outlet TCV" for the on service HX while performing action for a Clearance on Service Water.

What effect would this action have on Reactor Coolant temperature with no Operator action?

☐ **A** No effect on RCS temperature because the RBCCW Service Water HX outlet valves are throttled to prevent run out of the Service Water pumps.

☐ **B** No effect to RCS temperature because RBCCW flow to the SDC HX is throttled to minimize temperature transients on the RCS when SDC is in service.

☑ **C** RCS temperature would start to lower because RBCCW temperature will lower when Service Water flow to the RBCCW HX increases.

☐ **D** RCS temperature would start to rise because RBCCW temperature would rise when Service Water cooling is isolated to the RBCCW HX.

Question Misc. Info: MP2*LOIT SW, SDC, 2207, NRC-2016

Justification

"A" WRONG; The Service Water Valves are throttled to maintain RBCCW header temperature and fail open on a loss of power and for the conditions stated in the stem a raise in SW flow will reduce RBCCW header temperature.
Plausible; A loss of power to SW-8.1A fails open and SW-9A/B/C is throttled to the non-service RB HX for minimum flow during the winter.

"B" WRONG; While RBCCW flow does not change, SW header flow rises overcooling RBCCW thus cooling the RCS
Plausible; It is correct that the RBCCW flow is manually throttled to the SDC HX could cause a misunderstanding to the Examinee that temperature would not change.

"C" CORRECT; SI-306 and SI-657 at 50% coupled with stable RCS temperatures informs the Examinee that the loading on the Service Water System is nowhere near capacity, therefore de-energizing SW-8.1A to its LOCA position will raise the flowrate through the on service RBCCW HX therefore lowering RBCCW temperatures.

"D" WRONG; The TCV/SIAS SW outlet valve for the RBCCW HX fails open therefore SW header flow rises overcooling RBCCW thus cooling the RCS.
Plausible; The Examinee may think that the Temperature Control Valve for the RBCCW HX fails closed.

References

25203-26008 SH-02

Comments and Question Modification History

Added SW injection to stem and removed plural of temperature and HX

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

Number K1.08 **RO** 3.5* **SRO** 3.5* **CFR Link** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

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

Question #: **54** | **Question ID:** 2016025 | ☑ RO ☐ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 1 | ☑ Selected for Exam | **Origin:** New | ☐ Past NRC Exam?

The plant is at 100% Power when Instrument Air header pressure inside Containment went to 0 psi.

BOP reports 2-IA-27.1 “IA CTMT ISOL” is CLOSED

Which of the following sets of alarms would also occur as a result of a loss of Instrument Air inside Containment without Operator action over the next several minutes?

☐ **A** C02 “RCP CONTROL BLEED-OFF RELIEF FLOW HI”
C03 “A/B/C/D RCP BLEED-OFF TEMP HI”

☐ **B** C02/03 “PRI DRAIN TANK TEMP HI”
C03 “A/B/C/D RCP VAPOR SEAL PRESSURE HI”

☐ **C** C02 “LETDOWN FLOW LO”
C03 “A/B/C/D RCP BLEED-OFF FLOW HI”

☑ **D** C02 “LETDOWN PRESSURE HI/LO”
C03 “PZR PRESSURE SELECT CHANNEL DEVIATION HI/LO”

Question Misc. Info: MP2*LOIT IA, NRC-2016

Justification

A WRONG; C02/03 “A/B/C/D RCP BLEED-OFF RELIEF FLOW HI” RCP SEAL HDR pressure goes up when the CTMT isolation valve closes due to a loss of air causing bleedoff relief flow to rise, however actual bleedoff flow goes down due to the higher back pressure making Bleedoff Temperature High false.
C03 “A/B/C/D RCP BLEED-OFF TEMP HI” is also WRONG cooling flow from RBCCW does not change.
Plausible; CH-507 fails open on a loss of IA which is the alternate path for RCP bleedoff flow, to the PDT through the relief valve which is at a lower back pressure than the VCT causing a higher flow Examinee may think that bleedoff flow would increase causing temperature to also increase. BLEED-OFF temperature HI is plausible if the Examinee believes that that service header of RBCCW also isolates with a loss of Instrument Air.
B WRONG; C02/03 “PRI DRAIN TANK TEMP HI” Although RCP Bleedoff gets redirected to the PDT the average RCP Bleedoff temperature is 120°F but the alarm setpoint is 200°F therefore the alarm would not come in due to the added input from RCP Bleedoff.
C03 “A/B/C/D RCP VAPOR SEAL PRESSURE HI” is correct due to Containment Isolation for BLEED-OFF will fail closed on a loss of Instrument Air causing bleedoff pressure to rise to the relief valve setting then going to the PDT.
Plausible; Examinee may think that the alarm setpoint for the PDT is the same as the Quench tank which is set at 120°F and that the higher temperature bleed off flow which has been diverted to the PDT due to the CTMT Isolation valve failing closed.
C WRONG; C02 “LETDOWN FLOW LO” is correct due to the CTMT Isolation valve failing closed. C02/03 “A/B/C/D RCP BLEED-OFF FLOW HI” is wrong because RCP SEAL HDR pressure goes up when the CTMT isolation valve closes due to a loss of air diverting flow through the relief valve to the PDT.
Plausible; CH-507 fails open on a loss of IA which is the alternate path for RCP bleedoff flow, to the PDT which is at a lower back pressure than the VCT causing a higher flow examinee may not remember the relief valve in the flow path to the PDT.
D CORRECT; CH-515 and CH-516 fail closed isolating letdown. RCS pressure rises due to a loss of Letdown with charging still in service, raising PZR level, thumb rule 40 gpm charging flow, 65gallons/% PZR lvl, 1.62 mins/%PZR lvl, 15 psi/%PZR lvl, therefore 2.7 minutes to PZR PRESS DEVIATION Alarm.

References

ARP 2590B-212, ARP 2590B-031, AOP 2563

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Question #: **54** **Question ID: 2016025** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☐ **Lower Order?**

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

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

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

Question #: 55 | **Question ID:** 53327 | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

Rev. 8 | ☑ **Selected for Exam** | **Origin:** **Bank** | ☐ **Past NRC Exam?**

A plant heatup is currently in progress with the following condition:

- RCS Pressure 200 psia
- RCS THOT = 203 °F
- RCS TCOLD = 199 °F
- Heatup Rate 10 °F/hr
- Actual PZR Level 40%

Primary Plant Operator notes that Both 2-AC-6 and 2-AC-7 (CTMT Purge Exhaust Inboard/Outboard Isolation) indicates open (Red light energized).

Which one of the following actions must be completed?

☐ **A** Ensure either 2-AC-6 or 2-AC-7 (CTMT Purge Exhaust inboard/outboard Isolation) closed and sealed with its fuses removed within 1 hour.

☐ **B** Ensure either 2-AC-6 or 2-AC-7 (CTMT Purge Exhaust inboard/outboard Isolation) closed and sealed with its fuses removed prior to MODE 3.

☐ **C** Ensure both 2-AC-6 and 2-AC-7 (CTMT Purge Exhaust inboard/outboard Isolation) closed and sealed with their fuses removed prior to MODE 3.

☑ **D** Ensure both 2-AC-6 and 2-AC-7 (CTMT Purge Exhaust inboard/outboard Isolation) closed and sealed with their fuses removed within 1 hour.

Question Misc. Info: MP2*LORT*2224 [103 CCV-01-C 2892] (12/5/97) VENT, T.S., CTMT, 2314B, NRC-2016

Justification

A - WRONG; BOTH dampers must be closed and sealed within 1 hour due to Tech. Spec. 3.6.3.2 requirements.
Plausible; Examinee may assume the requirement for 1 hour action may be satisfied by ensuring the pathway is isolated by at least one damper.

B - WRONG; Tech. Spec. 3.6.3.2 has a one hour action statement that requires both dampers be closed and sealed in Mode 4 or above.
Plausible; Examinee may believe the governing TSAS is based on the requirements of CIAS (>/= Mode 3), which is the ESAS signal that would isolate CTMT on an accident, and CTMT Isolation, which requires the open isolation valve/damper be capable of being closed by an automatic signal.

C - WRONG; Tech. Spec. 3.6.3.2 applicability is Mode 4 or above.
Plausible; Examinee may believe the governing TSAS has the same requirements as CIAS (>/= Mode 3).

D - CORRECT; IAW OP-2314B (Rev. 21), Prerequisite 2.1.6, In Mode 4 or above, T.S. 3.6.3.2 requires AC-4, 5, 6, and 7 to be sealed closed with their fuses removed.

References

OP 2314B

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.2.39 **RO** 3.9 **SRO** 4.5 **CFR Link** (CFR: 43.2 / 45.13)

Knowledge of less than or equal to one hour Technical Specification action statements for systems.

Question #: **56** | **Question ID: 2016002** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

A plant startup is in progress following a refueling outage with reactor power presently at 25% with Group 7 CEAs at 150 steps.

The RO withdraws CEAs then releases the withdrawal switch but the switch contacts have stuck and Group 7 CEAs continues to withdraw.

Which of the following conditions will eventually stop CEA withdrawal based on the stuck switch contacts?

☑ **A** The high RCS boron concentration will allow reactor power to rise unabated until two High Power pretrips on RPS trigger and stop the CEA withdrawal.

☐ **B** The rising RCS Tcold will drive up the TM/LP trip setpoint calculation in RPS until 2 TM/LP pretrips trigger on RPS and stop the CEA withdrawal.

☐ **C** The increasing shift of reactor power to the top of the core, along with the rise in reactor power, will cause an early trigger of the Local Power Density trip.

☐ **D** The diminished CEA worth caused by the high RCS boron concentration will allow a continued CEA withdrawal until the Upper Core Stop triggers.

Question Misc. Info: MP2*LOIT, CED-01-C, 2556, NRC-2016

Justification

A - CORRECT; MTC will be very small at this point in core life, preventing an increasing RCS temperature from tempering the power rise. Reactor power only has to rise about 8% before High Power pretrips from RPS trigger, and stop the withdrawal by triggering a CEA Withdrawal Prohibit (CWP triggered by 2/4 Hi Pwr or TM/LP pretrips).

B - WRONG; At this power level the TM/LP setpoint is at its floor value, almost 400 psi below RCS pressure.
Plausible; At 100% power, this would possibly be the first pretrip to trigger, especially due to the added conservative effect on the TM/LP setpoint of ASI shifting more negative as CEAs withdraw.

C - WRONG; The only reason to withdraw CEAs for ASI control is if ASI were shifted to much toward the bottom of the core (too positive). This would put the LPD setpoint far out of reach of the shifting ASI as CEAs withdraw.
Plausible; If ASI were already negative, the added negative effect on ASI of withdrawing CEAs, compounded by the rise in power making the LPD setpoint more conservative, could combine to challenge this RPS trip setpoint.

D - WRONG; The full withdrawal of the first 64 CEAs was enough to raise reactor power about 6.5 decades. The last 9 CEAs only have to raise power about 8% to trigger a CWP that will stop CEA withdrawal.
Plausible; CEA worth is substantially diminished by the high boron concentration in the RCS and the CEA Upper Core Stop is only 27 steps above the initial CEA position at the start of the withdrawal.

References

ARP 2590C-110, OP 2203

Comments and Question Modification History

Changed stem to reduce the numbers of words and for clarity.

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

Number K3.02 **RO** 3.4* **SRO** 3.5 **CFR Link** (CFR: 41.7/45.6)

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

Question #: **57** **Question ID:** **170314** ☑ **RO** ☐ **SRO** ☐ Student Handout? ☐ Lower Order?

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

The plant is in normal operation at 100% power with the controlling pressurizer pressure controller setpoint set at 2250 psia.

The pressurizer level insurge bistable (which actuates ~3.6% above setpoint) has failed such that it will NOT actuate on a pressurizer insurge.

Then, a perturbation in a secondary system causes a pressurizer insurge that raises pressurizer level to 69% and causes a corresponding rise in pressurizer pressure.

What action will the Pressurizer Level and Pressure Control System take to automatically stop the rise in pressure when level rises to 69%?

☐ **A** Backup charging pumps running in Manual will stop and the spray valves will go full open.

☐ **B** Backup charging pumps running in Manual will stop and the spray valves will be partially open.

☑ **C** The proportional heaters will go to minimum output and the spray valves will be partially open.

☐ **D** The proportional heaters will remain at maximum and the spray valves will go full open.

Question Misc. Info: MP2 LOIT PLC-01-C MB-2325 2304A, PLPCS, NRC-2016

Justification

A - WRONG; At 50 psi above setpoint (2300 psia here), spray valves will start to open, and at 100 psi above setpoint (2350 psia) be full open. An insurge to 69% raises PZR level 4%, which equates to a pressure rise of about 60 psi. Therefore, PZR pressure would rise to about 2310 psia, which is high enough to open the spray valves about 20%.
Plausible; Chosen if the examinee uses the logic normally seen during training scenarios, where PZR sprays are being forced. If that were the condition, spray valves would respond immediately and be fully open with a 50 psi rise in pressure.

B - WRONG; The signal to stop the backup charging pumps manually started comes from the "+3.6%" bistable, which the stem has stated as failing in the NOT actuated mode. Therefore, there is no signal to stop the backup charging pumps on this insurge.
Plausible; Chosen if examinee confuses the +3.6% bistable affect on the charging pumps with the bistables that affect PZR heaters.

C - CORRECT; proportional heaters go to min. ~25psi above setpoint and the spray valves start opening at 50psi above setpoint.

D - WRONG; due to the failed bistable, the proportional heaters will not respond this way, but will respond only to pressure.
Plausible; If examinee assumes the failed bistable affects only the backup charging pump response.

References

OP-2204 Attachment 3;

Comments and Question Modification History

Added "TO" before 69 in stem.

NRC K/A System/E/A **System** 011 Pressurizer Level Control System (PZR LCS)

Number K3.03 **RO** 3.2 **SRO** 3.7 **CFR Link** (CFR: 41.7 / 45.6)

Knowledge of the effect that a loss or malfunction of the PZR LCS will have on the following: PZR PCS

Question #: **58** | **Question ID:** **8600105** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

Rev. 1 | ☑ **Selected for Exam** | **Origin:** **Bank** | ☐ **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 1 Thot input to the Reactor Regulating System suddenly fails to 1200 °F.

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 failed Loop 1 Thot has been automatically "bypassed" and calculated outputs are at pre-failure levels.

☐ **C** Select "Local-Setpoint" on the selected pressurizer level controller and ensure output is at the pre-failure level.

☐ **D** Transfer all steam dump controls to the Foxboro IA controller screen and ensure setpoints are at normal values.

Question Misc. Info: MP2*LOIT 2304, CVCS, PLPCS, PZR Level, Fox I/A, NRC-2008, NRC-2016

Justification

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. Plausible; Examinee may confuse actual controlling "brain" for the system as all functions are performed on an interface with the PPC.

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.

C - WRONG; This will PREVENT the RRS/Foxboro IA from controlling pressurizer level as designed in the event of a plant trip. Plausible;

D - WRONG; This action may be warranted if the Foxboro IA Tavg signal were to fail high with the failed input. However, when the steam dumps are transferred to the Foxboro IA, the auto setpoints for the two ADVs fails to 1200 psia and PIC-4216 fails to 985 psig, effectively PREVENTING the steam dumps from modulating, per design, in the event of a plant trip.
Plausible;

References

LP-RRS-00-C.R4C1

NO Comments or Question Modification History at this time.

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 / 45.5 to 45.8)

Ability to manually operate and/or monitor in the control room: NNI channel select controls

Question #: **59** | **Question ID:** **8100018** | ☑ **RO** ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

Rev. **2** | ☑ **Selected for Exam** | **Origin:** **Bank** | ☐ **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.
- 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 and what action must be taken to mitigate the effects of this condition?

☐ **A** Pressurizer pressure: 250 psia; CET Max: 405° F; CET High: 400°F
Reduce RCS pressure to raise LPSI flow.

☐ **B** Pressurizer pressure: 50 psia; CET Max: 285°F; CET High: 280°F
Start the Facility 2 Safety Injection Pumps.

☑ **C** Pressurizer pressure: 250 psia; CET Max:425°F; CET High: 420°F
Start the Facility 2 Safety Injection Pumps.

☐ **D** Pressurizer pressure: 50 psia; CET Max:415°F; CET High: 410°F
Reduce RCS pressure to raise LPSI flow.

Question Misc. Info: MP2*LOIT, ICCS, CET, SCM, 2387, MB-08084, NRC-2008 [K/A: 017/A2.02], NRC-2016 [Minor mod]
Requires use of Steam Tables

Justification

A - WRONG; RVLMS @ 0% only means the lowest RVLMS thermocouple (at 7%) is uncovered, 400° F is saturation for 250 psia.
Plausible; Examinee may feel conditions are superheated based on CET MAX and pressure must be lowered to allow the running LPSI pump to inject (it is close to shutoff head).

B - WRONG; RVLMS @ 0% only means the lowest RVLMS thermocouple (at 7%) is uncovered, P/T relationship is saturated for CET High.
Plausible; Examinee may feel conditions are superheated based on CET MAX.

C - CORRECT; With 250 psia, and CET High at 420°F, conditions indicate superheat at the top of the core. Superheat 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 either increasing the heat removal of the SGs (reflux cooling) or starting the Facility 2 Safety Injection Pumps, which failed to automatically start due to the loss of VA-20.

D - WRONG; Although the core is indicating uncovered, attempting to lower pressure in a saturated RCS is not a good option.
Plausible; Examinee may focus on the fact that P/T relationship is superheated by a large margin and want the higher capacity LPSI pumps in play to refill the core as fast as possible, as opposed to the lower flow HPSI pumps of the other facility.

References

steam table

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 017 In-Core Temperature Monitor System (ITM)

Number A4.02 **RO** 3.8 **SRO** 4.1 **CFR Link** (CFR: 41.7 / 45.5 to 45.8)

Ability to manually operate and/or monitor in the control room: Temperature values used to determine RCS/RCP operation during inadequate core cooling (i.e., if applicable, average of five highest values)

Question #: **60** | **Question ID: 2016003** | ☑ **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 functioning as designed.

Then, a loss of DV-10 occurs, resulting in a plant trip. During the performance of EOP 2525, Standard Post Trip Actions, the operators noted several abnormal, post-trip indications.

Which one of the following describes the cause and effect that the loss of DV-10 had on plant systems and components?

☑ **A** Both MSIVs failed closed, causing a sudden spike in SG pressure, which in turn caused SG water level to shrink below the RPS low level trip setpoint.

☐ **B** Both MSIVs failed closed, causing a sudden spike in RCS temperature, resulting in the TM/LP trip setpoint being driven into the existing RCS pressure.

☐ **C** The Main Turbine instantly tripped on loss of two turbine undervoltage relays, causing a sudden spike in SG pressure, which in turn caused SG water level to shrink below the RPS low level trip setpoint.

☐ **D** The Main Turbine instantly tripped on loss of two turbine undervoltage relays, causing a sudden spike in RCS temperature, resulting in the TM/LP trip setpoint being driven into the existing RCS pressure.

Question Misc. Info: MP2*LOIT, MSIVs, Load Reject, NRC-2016

Justification

A - CORRECT; DV-10 powers the Facility 1 solenoids in each MSIV, that are designed to be deenergized by ESAS to close the MSIVs on an actuation signal. Therefore, the loss of DV-10 fails both MSIVs to their closed, accident position. This results in an instantaneous total load reject that was not the result of a turbine trip, which causes SG pressure to spike well above the MSSV setpoints. Because a trip has not yet been processed, the steam dumps will open only on a rise in SG pressure, which adds to the sudden SG pressure rise. Because of this compounding effect on SG pressure rise, RPS often processes the first trip signal on low SG water level, which is caused by level shrinkage on the rising SG pressure.

B - WRONG; The TM/LP trip setpoint would rise very quickly, but because RCS pressure is also going up, it would not catch it before the high RCS pressure trip or the low SG level trip.
Plausible; Examinee may remember that the TM/LP trip setopint rises over 14 psi for every degree rise in Tcold, which would appear to make it possible for the setpoint to be driven up the required amount (about 150 psi) to hit the normal RCS pressure. They may also confuse this trip with the High RCS pressure trip, which would actuate in a load reject situation.

C - WRONG; Loss of DV-10 will instantly close both MSIV, which will immediately drive SG pressure high and shrink both SGs. The Main Turbine will trip on a the reactor trip, not the loss of DV-10 directly.
Plausible; Examinee may remember that on a load reject, the plant has a 50-50 chance of tripping on either high RCS pressure or Low SG Water Level.

D - WRONG; The main turbine undervoltage relays will not deenergize on the loss of DV-10 directly, but all four will deenergize on the reactor trip.
Plausible; Examinee may remember that a load reject (main turbine trip) at 100% power would cause the RCS to trip on high pressure, due to the sudden and dramatic rise in RCS temperature, and confuse that effect with the one stated.

References

AOP 2506A

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 035 Steam Generator System (S/GS)

Number K6.01 **RO** 3.2 **SRO** 3.6 **CFR Link** (CFR: 41.7 / 45.7)

Knowledge of the effect of a loss or malfunction on the following will have on the S/GS: MSIVs

Question #: **61** | **Question ID:** 1681567 | ☑ RO | ☐ SRO | ☐ Student Handout? | ☐ Lower Order?

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

The plant tripped from 100% power when the #2 MSIV closed on failure of the valve operator.

During the performance of EOP 2525, Standard Post Trip Actions, the following conditions exist:

- RCS Tc 532 °F and stable.
- Steam Generator levels 35% and slowly trending up.
- Steam Generator pressures are 915 psia and stable.
- Annunciator DB-17 on C-05, "STM GEN NO. 2 SAFETY RELIEF VLVE OPEN," is in alarm.
- All other plant systems and components are responding as designed.

What actions does the BOP take in EOP 2525 in response to alarm DB-17 on C-05?

☐ **A** Reduce S/G pressure to approximately 840 psig using the 'A' condenser steam dump.

☑ **B** Reduce S/G pressure to approximately 880 psia using the #2 Atmospheric Dump Valve.

☐ **C** Reduce RCS Tavg to 530 °F - 535 °F using the TIC-4165, RCS Temperature Controller.

☐ **D** Close #1 MSIV and reduce SG pressure to </= 880 psia using the Atmospheric Dump Valve.

Question Misc. Info: MP2*2525, 2260, Loss of VR-11

Justification

A - WRONG; The "A" steam dump units are in psig and setting the controller to 840 psig equates to 855 psia at the valve, but due to main steam header pressure losses would amount to about 880 psia in the S/G. Although this is the pressure needed to ensure the lowerest set MSSV is not stuck open, the condenser steam dumps whould have to lower RCS temperature much further than the #2 ADV, because they can only lower the #2 S/G pressure by cooling the RCS down enough to pull energy from the S/G.
Plausible; Examinee may consider using 'A' steam dump valve as it would be easier to control both SGs using one valve.

B - CORRECT; Per EOP 2525: S/G pressure band is 880 - 920. However, the lowest MSSV setpoint is ~ 1000 psia, with a possible blowdown value of about 100 psi. Therefore, to verify a MSSV is not stuck open, OP 2260 directs the ADVs be adjusted to lower S/G pressure to the "lower end of the control band", or 880 psia. This would equate to an RCS Tc of about 530 °F.

C - WRONG; RCPs are running, so this Tavg band could equate to a steam pressure over 900 psia, which is potentially above the lowest set MSSV blowdown floor value. OP 2260 gives guidance to reduce SG pressure to the "low end of the band" (IAW EOP 2525, that's 880 psia), to ensure all MSSVs reclose.
Plausible; The examinee may consider the expectation to restore Tavg to a normal, post-trip level.

D - WRONG; Although this would ensure even steam demand of both S/Gs, it is no longer required, provided the condenser can be maintained as a heat sink.
Plausible; This is the expected action for a trip with the loss of the main condenser, where RCS temperature would then be controlled by the ADVs.

References

OP 2260

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

Number K5.01 **RO** 2.9 **SRO** 3.2 **CFR Link** (CFR: 41.5 / 45.7)

Knowledge of the operational implications of the following concepts as the apply to the SDS: Relationship of no-load T-ave. to saturation pressure relief setting on valves

Question #: **62** **Question ID:** **56581** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☑ **Lower Order?**

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

An Aerated Radwaste Discharge is in progress using the normal system configuration.

Which of the following is an indication that the Aerated Radwaste Discharge Valves have been accidentally OVERRIDDEN open for this discharge?

☑ **A** The "Hi Rad/Inst. Fail" annunciator is alarming and the Aerated Radwaste Discharge Final Filter delta-P is reading 15 psig.

☐ **B** The "Hi Rad/Inst. Fail" annunciator is alarming and the Aerated Radwaste Monitor Pump is still running.

☐ **C** Aerated Waste Monitor tank pump recirculation valve LRA-55.1 remians open during the discharge.

☐ **D** A normal discharge flow rate to Long Island Sound and the Aerated Radwaste Monitor tank level is going up.

Question Misc. Info: MP2*LOIT PIO-04-C, 2617A, NRC-2000, NRC-2016

Justification

A - CORRECT; If the "Hi Rad/Inst. Fail" annunciator is alarming indicates the rad monitor has failed or is in alarm and has triggered the interlock to close the discharge valves. However, if the Aerated Radwaste Discharge Final Filter delta P is reading 15 psig, then there must still be flow out the discharge valves. Therefore, the valves must be overridden open.

B - WRONG; The rad monitor has triggered a discharge isolation signal but this has no impact on the monitor tank pump.
Plausible; The monitor tank pump does auto trip on a low level in the tank, which on occasion has caused the rad monitor to alarm.

C - WRONG; Athough the recirc valve is open at the start of the discharge, there are no interlocks with this valve and the discharge valves.
Plausible; Examinee may recall from the discharge procedure that the recirc valve is open at the start of the discharge and may think that the recirc valve goes closed when the discharge valves go open therefore attribute an interlock between the valves.

D - WRONG; A rise in tank level, as opposed to a decrease, is not indicative of overridden discharge valves. There is no auto trip for this condition.
Plausible; This would invalidate the permit, triggering a violation of state and federal requirements for a radioactive discharge to Long Island Sound.

References

ARP 2590H-025, SP-2617A

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 068 Liquid Radwaste System (LRS)

Number K6.10 **RO** 2.5 **SRO** 2.9 **CFR Link** (CFR: 41.7 / 45.7)

Knowledge of the effect of a loss or malfunction on the following will have on the Liquid Radwaste System: Radiation monitors

Question #: **63** **Question ID:** **1100009** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☑ **Lower Order?**

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

A waste gas discharge is in progress. In addition to the Waste Gas Discharge Radiation Monitor, RM-9095, which of the following ventilation radiation monitor(s) will also show a change in the waste gas discharge activity level due to the normal flowpath of the discharge?

☐ **A** Unit 2 Stack Radiation Monitors, RM-8132A and B

☑ **B** Millstone Stack Wide Range Gas Monitoring System, RM-8169

☐ **C** Unit 2 Kaman Ventilation Radiation Monitor, RM-8168

☐ **D** Aux. Bldg. -25' Rad. Waste Ventilation Radiation Monitor, RM-7896.

Question Misc. Info: MP2*LOIT [071 GRW-04-C] 2337, GRWS, NRC-2016

Justification

"A" - WRONG; A waste gas discharge does NOT discharge through the Unit 2 Stack; therefore the Unit 2 Stack Radiation Monitors, RM-8132A and B are NOT used.
Plausible; Examinee may believe the waste gas discharge must go out the stack of the unit making the discharge to ensure proper control and accountability.

"B" - CORRECT; The waste gas discharge flow path is through the Waste Gas Discharge Radiation Monitor, RM-9095, and out the Millstone Stack which utilizes the Millstone Stack Wide Range Gas Monitoring System, RM-8169.

"C" - WRONG; A waste gas discharge does NOT discharge through the Unit 2 Stack; therefore the Unit 2 Kaman Ventilation Radiation Monitor, RM-8168 is NOT used.
Plausible; Examinee may feel it logical that because the Kaman is used for monitoring the potentially very high rads discharged from an accident (units of micro-curies instead of "mr" or "cpm") that it must be in line to see the slightly above normal rads discharged with a waste gas discharge.

"D" - WRONG; Although the a waste gas discharge originates on the -25 ft level of the Aux Building, it does not utilize the ventilation system flow path that goes past RM-7896.
Plausible; Examinee may believe the discharge would use the same rad monitor that is used to measure radioactive gasses emanating from that area of the plant.

References

SP 2617BE

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 071 Waste Gas Disposal System (WGDS)

Number A1.06 **RO** 2.5 **SRO** 2.8 **CFR Link** (CFR: 41.5 / 45.5)

Ability to predict and/or monitor changes in parameters(to prevent exceeding design limits) associated with Waste Gas Disposal System operating the controls including: Ventilation system

Question #: **64** | **Question ID:** **8056807** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**
Rev. **2** | ☑ **Selected for Exam** | **Origin:** **Bank** | ☐ **Past NRC Exam?**

The plant is in Mode 4, starting up, following a 30 day refueling outage. Fuel is being moved in the Spent Fuel Pool (SFP).

While an ATI alarm is being investigated on ESAS, an inadvertent EBFAS actuation is triggered.

What affect would this have on Ventilation System in the SFP area?

☐ **A** AEAS ventilation realignment is only possible using the local push buttons.

☑ **B** The EBFAS actuation would prevent AEAS realignment of SFP ventilation.

☐ **C** SFP area Air Conditioner Unit isolates, suspend work if general area >100°F.

☐ **D** Fresh air makeup damper closes, all work on SFP floor must be suspended.

Question Misc. Info: MP2*LOIT*4262 [013 ESA-01-C 995] (8/27/96) 2384, ESAS, APP, NRC-2008 (K/A; 072-K3.02), NRC-2016

Justification

A - WRONG; Because EBFAS has triggered, AEAS is blocked, even if local push buttons are pushed.
Plausible; Examinee may logically think that a high radiation in the SFP area (AEAS trigger and outside CTMT) would have ventilation alignment priority over an event inside containment (EBFAS trigger), which is most probably contained by the CTMT barrier.

B - CORRECT; EBFAS being triggered would no longer realign the SFP ventilation system because the new "Source Term" analysis has made it unnecessary for AEAS to be in operation for fuel movement. Fuel movement in the SFP area may continue provided all boundaries remain intact.

C - WRONG; Although the function of AEAS is to prevent the release of fission product gasses in the event of a fuel handling accident, it does not isolate the A/C unit.
Plausible; Examinee may believe that the AEAS signal isolates the A/C Unit since it non vital and therefore necessity for suspending fuel operations if the local area temperature exceeds 100°F would be correct.

D - WRONG; EBFAS does not isolate the fresh air makeup to the SFP, AEAS does.
Plausible; Examinee may confuse AEAS isolation of fresh air makeup with EBFAS actuation which would require work to be suspended due to no fresh air.

References

SP 2609I

Comments and Question Modification History

Made modifications to the question and answers to remove the possibility of a SRO question only requirements.

NRC K/A System/E/A **System** 072 Area Radiation Monitoring (ARM) System

Number A3.01 **RO** 2.9* **SRO** 3.1 **CFR Link** (CFR: 41.7 / 45.5)

Ability to monitor automatic operation of the ARM sys- tem, including: Changes in ventilation alignment

Question #: **65** **Question ID: 1689529** ☑ **RO** ☐ **SRO** ☐ **Student Handout?** ☐ **Lower Order?**

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

The plant is in normal operation at 100% power, when a Fire System Trouble annunciator is received on C06/7 and Zone 45 on Fire Panel, C-26. The Auxiliary Building PEO subsequently calls from the West DC Switchgear Room and reports the following:

- One Ion Chamber smoke detector is in alarm.
- The Halon strobe lights and horn are pulsating slowly.
- All other smoke detectors are operating normally (not in alarm).
- There is no smoke or fire in the area.
- The detector appears to have failed.

Which of the following describes the impact of the above conditions, and a required action per the applicable ARP?

☐ **A** The Fire Suppression system is alarming as a warning of a potential for a discharge due to a system malfunction. Place the panel in override and have the Auxiliary Building PEO increase monitoring of the room.

☑ **B** The Fire Suppression system is alarming as a warning of a potential for a discharge. Disable the halon system to prevent an inadvertant discharge and provide backup fire suppression and dedicated fire watch to monitor the room.

☐ **C** The Fire Suppression system is warning that a discharge will occur after a timer countdown. Place the panel in override to prevent the impending discharge and have the Auxiliary Building PEO increase monitoring of the room.

☐ **D** The Fire Suppression system is warning that a discharge will occur after a timer countdown. Once the room has been ventilated of halon, provide backup fire suppression and establish a fire watch to monitor the room.

Question Misc. Info: MP2*LOUT, FPS-01-C, Fire, Halon, DC Swgr, NRC-2009 (SRO), NRC-2016

Justification

A - WRONG; The Halon system is not in a state of potential discharge because the detection system has a failure. Therefore, there is no need to manually override the system so it can't function at all.
Plausible; Examinee may think that due to the "false" activation of a sensor, the system should be prevented from any subsequent activation and the Halon system can no longer trigger.

B - CORRECT; The East and West DC switchgear rooms require two zones (one photoelectric smoke detector and one ion smoke detector) to initiate a halon release. Activation of one smoke detector zone, ion or photoelectric, will cause the strobe and horn to pulse slowly. However, the admin requirements state all detectors must be functioning or the system is inoperable and a backup means must be initiated.

C - WRONG; Activation of a second smoke detector of the opposite type, but in the same room, will cause the strobe and horns for the affected room to pulse QUICKLY. The flashing lights will operate, and a 60 second pre-discharge time delay will begin. Upon expiration of the time delay the Halon System will discharge and the strobe and horn will sound steadily.
Plausible; Examinee may think that the SLOWLY pulsating horn and strobe light warn of a timer countdown to discharge halon, in which case, the Halon system would then be inoperable and this action would be correct.

D - WRONG; Only one detector failing in the activate mode would cause the given alarms.
Plausible; Examinee may think that the pulsating horn and strobe lights indicate that the failed detector has caused a full system malfunction and a discharge is imminent. If the system were actually triggered due to multiple detector failures, this would be the correct choice.

References

TRM 3.3.3.7, ARP 2590I, TRM 3.7.9.4

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 086 Fire Protection System (FPS)

Number A2.01 **RO** 2.9 **SRO** 3.1 **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Ability to (a) predict the impacts of the following mal- functions or operations on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Manual shutdown of the FPS

Question #: **66** | **Question ID:** **1689801** | ☑ **RO** | ☐ **SRO** | ☑ **Student Handout?** | ☐ **Lower Order?**
Rev. **0** | ☑ **Selected for Exam** | **Origin:** **Bank** | ☐ **Past NRC Exam?**

The plant tripped from 100% power due to a Large Break Loss Of Coolant Accident and the crew is now progressing through EOP 2532, Loss of Coolant Accident.

The following conditions existed approximately 15 hours after the trip:

- VR-11 is de-energized.
- RWST level = ~ 8% and stable.
- RCS pressure = ~ 35 psia and dropping very slowly.
- CTMT pressure = ~ 2 psig and dropping very slowly.
- All plant equipment is functioning as designed.
- All applicable procedure steps are either progressing as required or have been successfully completed.

Then, SI Pump amps and flow begin to fluctuate rapidly.

IAW EOP 2541, App 2, Minimum SI Flow Curves, what is the minimum required SI flowrate at this time?

☐ **A** ~ 65 gpm

☑ **B** ~ 130 gpm

☐ **C** ~ 570 gpm

☐ **D** ~ 3200 gpm

Question Misc. Info: MP2*LORT 2540, 2540C1, CTMT Sump Clogging, SI Flow Requirements

Justification

A - WRONG; 65 gpm is the flow rate on Figure 5 that corresponds to half the flow required by the curve, NOT the actual required flow.
Plausible; Examinee may divide the flow in half thinking that the flow indicated should be doubled based on loss of VR-11.

B - CORRECT; Conditions are indicative of CTMT sump clogging. EOP 2532, "LOCA", directs to throttle HPSI flow to maintain minimum ECCS flow for decay heat removal, as specified in Appendix2, Figure 5. 15 hours=900 minutes. Per this curve 900 minutes corresponds to approximately 130 gpm.

C - WRONG; 570 gpm corresponds to the RCS pressure of 35 psia on Figure #4, normal post-SRAS conditions. However, this is NOT the correct curve to use for "clogged CTMT sump" conditions.
Plausible; Examinee may use this curve because it is the one that would normal be used at this time.

D - WRONG; 3200 gpm corresponds to the RCS pressure of 35 psia on Figure #3, normal pre-SRAS conditions. However, this is NOT the correct curve to use for "clogged CTMT sump" conditions.
Plausible; Examinee may not have realized the RWST level is not gong down, indicative of a post-SRAS condition, or the required flow rate if CTMT spray is still in service. However, at this CTMT pressure, CTMT spray would have been secured.

References **Provided**

Provided During Exam: EOP 2541, App. 2, Figures, SI Flow Curves.

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 2.1.25 **RO** 3.9 **SRO** 4.2 **CFR Link** (CFR: 41.10 / 43.5 / 45.12)

Ability to interpret reference materials, such as graphs, curves, tables, etc.

Question #: **67** | ***Question ID:*** ***8000064*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

Unit 2 is operating at 100% power with shift turnover in progress.

- Four PEOs have arrived to relieve the watch.
- Two of the on-coming PEOs are qualified as Fire Brigade Members.
- One of these two members is qualified as Fire Brigade Leader.
- All PEOs are fully qualified watch standers.

The PEOs sign in to cover the following positions:

- 1 Fire Brigade position
- Tech. Spec.
- TRM Appendix 'R'

After approximately 2 hours, the Unit 2 Fire Brigade Leader qualified PEO must go home due to an illness.

Which of the following describes the actions required to meet the administrative minimum crew manning?

☐ **A** Have the other fire Brigade qualified PEO sign in as Tech. Spec. and Fire Brigade member positions.

☐ **B** Have the other fire Brigade qualified PEO sign in as TRM Appendix 'R' and Fire Brigade member positions.

☑ **C** Have the Unit 2 Work Control SRO sign in as the TRM Appendix 'R' operator then the spare operator assumes the Fire Brigade position.

☐ **D** Have the on-shift BOP sign in as the TRM Appendix 'R' operator then the spare operator assumes the Fire Brigade position.

Question Misc. Info: MP2, LOIT, Fire, TRM, NRC-2016

Justification

A - WRONG; The Tech. Spec. position cannot be covered by any other required position (no double-duties), including Fire Brigade. Plausible; Examinee may consider covering both positions with one PEO as not being a conflict because one is covered by Tech. Specs. and one is covered by the TRM (which is not part of the license).

B - WRONG; The App. 'R' position cannot be covered by any other required position (no double-duties), including Fire Brigade. Plausible; Examinee may consider the App. 'R' position and Fire Brigade position as effectively the same thing as both deal with a fire.

C - CORRECT; TRM 6.2.2 states that the fire brigade will consist of at least 5 members (from Unit 2 and Unit 3) and shall NOT include two members of the shift crew necessary for the safe shutdown of the unit or the App. 'R' designated operator. Although 2 PEOs are fire brigade qualified, only 1 of them is available for the fire brigade; 2 PEOs must be designated as Tech. Spec. and one must be designated as App. 'R'. Also, the App. 'R' position cannot be covered by any other required position (no double-duties) and the BOP is one of the two required RO positions in the control room. Therefore, only a spare operator (WC SRO) qualified as a PEO or higher can be used to replace the PEO who went home sick.

D - WRONG; The App. 'R' position cannot be covered by any other required position (no double-duties) and the BOP is one of the two required RO positions in the control room. Therefore, only a spare operator (WC SRO) qualified as a PEO or higher can be used to replace the PEO who went home sick.
Plausible; Examinee may think a fully qualified RO can maintain the Tech. Spec. manning requirements and App. 'R' requirements as the control room RO's are required to perform App. 'R' actions outside the control room during an App. 'R' fire.

References

TRM 6.2.2

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 2.1.5 **RO** 2.9 **SRO** 3.9 **CFR Link** (CFR: 41.10 / 43.5 / 45.12)

Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

Question #: **68** | **Question ID: 8055096** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**
Rev. 2 | ☑ **Selected for Exam** | **Origin: Bank** | ☐ **Past NRC Exam?**

The plant is in Mode 4, preparing to enter a refueling outage. Cooldown and depressurizing of the RCS is on hold for equipment testing before Mode 5 entry.

The following conditions now exist:

- RCS pressure is stable at 150 psia.
- RCS Tavg is stable at 225 °F
- Pressurizer (PZR) level is 40%, as seen on the Cold Cal. Indication L-103.
- Channel "X" and "Y" PZR level indicate 46% on both controllers.
- ALL PZR Backup Heaters are in "PULL-TO-LOCK".

Which one of the following control actions is required to maintain PZR level and pressure stable?

☐ **A** Level controller in AUTO with Local/Remote switch in LOCAL and setpoint set to 40%.
Pressure controller in MANUAL with output adjusted as necessary.

☑ **B** Level controller in MANUAL with output adjusted as necessary.
Proportional Heater Breakers opened and closed as necessary.

☐ **C** Foxboro IA Level Setpoint MANUALLY set to 46%.
Proportional Heater Breakers opened and closed as necessary.

☐ **D** Level controller in AUTO set to 40% with Setpoint Override switch (C-02) in OVERRIDE.
Pressure controller in MANUAL with output adjusted as necessary.

Question Misc. Info: MP2*LOIT*2457 [010 PLC-01-C 4819] (9/15/97) 2304A, PLPCS, NRC, APP, NRC-2008

Justification

A - WRONG; The "insurge" relay that causes the proportional heaters to be at maximum output does not receive any input from the level controllers and the pressure controllers are still unable to control Proportional Heaters.
Plausible; Examinee may focus on the fact that this would bypass the RRS level setpoint (of 40%) and allow automatic control of PZR level. This is an option for level control because it bypasses the decalibrated level setpoint, but would not restore pressure control.

B - CORRECT; With RCS temperature in the range for SDC operation, the Reactor Regulating System (RRS) 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 ~46%. 46%-40%=6% 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.

C - WRONG; The RRS is a subsystem of the Foxboro IA computer system. The Foxboro IA generates the PZR level setpoint signal that is used by the level control circuit, using RCS Tc and Th inputs. The only manual control for PZR level setpoint generation is which RCS loop is used for temperature input.
Plausible; Examinee may expect this is possible that five other setpoints generated by the Foxboro IA can be manually overridden.

D - WRONG; However, it does NOT override the insurge signal that drives the proportional heaters to maximum output, therefore the pressure controllers are still unable to control Proportional Heaters.
Plausible; Examinee may recognize that placing the PZR Setpoint Override Switch in the "OVERRIDE" position will override the insurge signal that prevents the backup charging pumps from running if manually started, and believe it also blocks the insurge signal to the PZR heaters.

References

OP 2207

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 2.1.31 **RO** 4.6 **SRO** 4.3 **CFR Link** (CFR: 41.10 / 45.12)

"Ability to locate control room switches, controls and indications and to determine that they are correctly reflecting the desired plant lineup."

Question #: **69** | **Question ID:** **86455** | ☑ **RO** ☐ **SRO** | ☐ Student Handout? | ☑ Lower Order?
Rev. 2 | ☑ Selected for Exam | **Origin:** **Bank** | ☐ Past NRC Exam?

The following plant conditions exist:
- Reactor power is at 91%
- Xenon is stable

Turbine Load Set Controls:
- Load Setpoint is 85.2%
- Load Demand is 85.2%
- Load Ramp Rate Setpoint is 5%/hr
- Load Hold is selected

The US then directs that the power ascension be continued. The BOP then selects "Load Resume" and then presses the Load Setpt "Raise" button.

Which of the following describes the response to selecting Load Setpt "Raise"?

☐ **A** Load Setpoint will immediately change to 86.2% and Load Demand will rise to this level at 5%/hr.

☐ **B** Load Setpoint will immediately change to 90.2% and Load Demand will rise to this level over the next hour.

☐ **C** A pop-up confirmation box asks; "Raise Turbine Load 1.0%?", with buttons for "OK" and "Cancel".

☑ **D** Load Setpoint will immediately change to 85.3% and Load Demand will rise to this level at 5%/hr.

Question Misc. Info: MP2 LOIT MTC Load Set Controls

Justification

A - WRONG; The "Raise" and "Lower" buttons change the setpoint by 0.1%, not 1.0%
Plausible; Examinee may have the magnitude of setpoint change reversed.

B - WRONG; The magnitude of setpoint change driven by the "Raise" and "Lower" buttons is not affected by the "Rate" setpoint.
Plausible; Examinee may believe the "Ramp Rate" setpoint is the rate at which Load Setpoint changes.

C - WRONG; The "confirmation box" only appears if a new setpoint value is directly keyed into the setpoint field.
Plausible; Examinee may believe the system will ask for load change confirmation regardless of how it is input.

D - CORRECT; "Raise" raises load setpoint by 0.1%. Due to their relatively small effect, the Raise and Lower "buttons" do not have pop-up confirmation windows for these actions. The load control function then changes the Load Demand at the selected loading rate (5%/hr - slowly) towards Load Setpoint (target).

References

OP 2204

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 2.2.2 **RO** 4.6 **SRO** 4.1 **CFR Link** (CFR: 41.6 / 41.7 / 45.2)

Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.

Question #: **70** | **Question ID:** **4006200** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

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

The plant has just completed a refueling outage with final system alignments being made.

The crew is in OP 2202A, "Reactor Startup by Dilution ICCE", performing Section 4 "Establish Initial Conditions for Reactor Startup".

The RO just finished diluting the RCS to get it within 150 ppm of the required boron concentration.

One minute after dilution flow has been stopped, all four channels of Wide Range NI's are slowly rising at an increasing rate.

Which of the following actions is the RO immediately responsible to perform?

☐ **A** Insert Group 7 CEAs until startup rate remains negative.

☑ **B** Commence emergency boration from the RWST or BAST.

☐ **C** Request the US reduce steam loads to raise RCS Tavg.

☐ **D** Notify HP and Sound the Containment evacuation alarm.

Question Misc. Info: MP2*LOIT NI, 2301E, 2304, 2558, AOP, LOIT, NRC-2016

Justification

A - WRONG; Per procedure, the RCS boron concentration is adjusted before regulating CEAs are withdrawn.
Plausible; The examinee may recognize SDM is not being met and CEA insertion is a prescribed action.

B - CORRECT; OP 2202A, "Precautions" direct CEA insertion or boron addition be performed whenever any abnormal behavior of the reactor is observed. Also, AOP-2558, Emergency Boration, entry conditions requires emergency boration when an unexplained rise in count rate is encountered. A sustained positive SUR should not be observed at this time, as all CEAs are still inserted and the boron concentration should sill meet SDM.

C - WRONG; This action is will not help in this instance due to the positive MTC that would be present.
Plausible; Examinee may recall that during a normal Pull-To-Critical operation, changing RCS temperatures due to automatic operation of secondary system components can impact the reactor startup.

D - WRONG; This is not the ROs responsibility at this time. The US would either perform this action or delegate it to another control room operator.
Plausible; Examinee may feel this is the RO's responsibility because the RO is the first to recognize the changing radiation levels in containment and it is an action that should be taken for personnel safety.

References

OP 2202A

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

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

Question #: **71** | ***Question ID:*** ***8500016*** | ☑ ***RO*** | ☐ ***SRO*** | ☐ **Student Handout?** | ☑ **Lower Order?**

Rev. **2** | ☑ **Selected for Exam** | ***Origin:*** ***Bank*** | ☐ **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 been granted per the Emergency Exposure Limits guidelines and documented on a Radiation Work Permit that covers the applicable high radiation area.

Which of the following exposure requirements would still be applicable for the individual (with no prior exposure for the year) who enters the high radiation area to assist the injured operator?

☐ **A** Any male or non-pregnant female can volunteer; stay time is limited to 6 minutes.

☐ **B** Only males over 50 can volunteer; stay time limit is 25 minutes.

☐ **C** Only males any age can volunteer; stay time is up to that individual.

☑ **D** Any male or non-pregnant female can volunteer; stay time is up to that individual.

Question Misc. Info: MP2*LOIT Emerg Rad Exposure, ALARA, NRC-2008 [K/A; 2.3.4], NRC-2016

Justification

A - WRONG; This equates to a dose of 5 rem, which is the normal limit for non-emergency scenarios.
Plausible; Examinee may recognize that any employee can volunteer, but use the normal exposure limits to calculate the stay time.

B - WRONG; This equates to a dose of 25 Rem, which is the Emergency Exposure Limit for non-volunteers performing accident mitigation. The "male over 50" is a company guideline when soliciting volunteers for high exposure missions, but it is NOT a requirement.
Plausible; Examinee may use the "accident mitigation" exposure limit as the highest allowed for any circumstance.

C - WRONG; This is the correct dose for life-threatening emergency situations. However, although excluding females is plausible, it is NOT an administrative requirement.
Plausible; Examinee may consider the "male over 50" company guideline as an actual requirement.

D - CORRECT; For "life saving situations" the dose limit per Emergency Exposure Limits is strictly up to the individual who volunteers to give assistance. In this instance, procedures do NOT have different requirements for males or females as the person must be a volunteer.

References

EPI-FAP09

Comments and Question Modification History

Added "non-pregnant" before female

NRC K/A System/E/A **System** 2.3 Radiation Control

Generic K/A Selected

NRC K/A Generic **System** 2.3 Radiation Control

Number 2.3.7 **RO** 3.5 **SRO** 3.6 **CFR Link** (CFR: 41.12 / 45.10)

Ability to comply with radiation work permit requirements during normal or abnormal conditions.

Question #: **72** **Question ID: 5000018** ☑ **RO** ☐ **SRO** ☐ Student Handout? ☑ Lower Order?

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

A steam generator tube rupture has occurred on SG2. EOP-2534, "Steam Generator Tube Rupture" has been implemented.

Which of the following actions is performed in accordance with EOP-2534, "Steam Generator Tube Rupture" to DIRECTLY limit the potential radiation release to the public?

☐ **A** Maintain ruptured Steam Generator level below 40% after SG isolation.

☑ **B** Ensuring ruptured SG ADV setpoint at 920 psia and closed after SG isolation.

☐ **C** Tripping RCPs if pressurizer press less that 1714 psia and SIAS initiated.

☐ **D** Entering EOP-2536 (ESDE) for a SG pressure < 800 psia and subcooling going up.

Question Misc. Info: MP2*LOIT 2534, SGTR, NRC-2005, NRC-2016 [Corrected Justification for Choice "A"]

Justification

CHOICE (A) - NO WRONG: Use of the TDAFP would cause an unmonitored release of contaminated steam to the environment.
VALID DISTRACTOR: the procedure directs a level band of 40% - 45% to allow for scrubbing of iodine from the RCS leakage into the S/G before it is released to the environment during the initial RCS cooldown phase (top of tube bundle is ~ 33%).

CHOICE (B) - YES The ADV is ensured to be in auto and its setpoint is raised to a value below the upper end of the band. It is also ensured to be closed since steam pressure should be below this point. This places the ADV in a condition to open prior to pressure in the isolated steam generator reaching the MSSV lift setpoint, and minimizing the possibility that a MSSV will open and stick in an open position.

CHOICE (C) - NO WRONG: RCP trip strategy based on worst-case LOCA concerns. Continued operation of pumps is preferable during a SGTR event to allow for a prompt controlled RCS cooldown and depressurization.
VALID DISTRACTOR: because RCP trip is directed by procedure under specified conditions.

CHOICE (D) - NO WRONG: Functional Recovery Procedure is used to address multiple events from a symptom-based perspective. EOP-2536 should not be entered and implemented from EOP-2534 with multiple events in progress
VALID DISTRACTOR: because EOP-2534 does contain diagnosis confirmation steps and the functional recovery does address excess steam demand events.

References

EOP 2534

Comments and Question Modification History

Added "after SG isolation" in answers "A" and "B"

NRC K/A System/E/A **System** 2.3 Radiation Control

Generic K/A Selected

NRC K/A Generic **System** 2.3 Radiation Control

Number 2.3.11 **RO** 3.8 **SRO** 4.3 **CFR Link** (CFR: 41.11 / 43.4 / 45.10)

Ability to control radiation releases.

Question #: **73** | **Question ID:** **2016006** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

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

Which one of the following is the RO's responsibility to monitor and report for the purpose of Event Classification during the applicable emergency event?

☐ **A** During a Steam Generator Tube Rupture, wind direction has changed while performing the initial cooldown.

☐ **B** During an Excess Steam Demand event, 200 °F subcooling was exceeded before the affected SG blew dry.

☑ **C** During a Small-Break Loss Of Coolant Accident, the ICC reactor head level has just dropped from 7% to 0%.

☐ **D** While performing EOP 2528, Loss of Offsite Power, the switchyard breakers for the last two offsite lines trip.

Question Misc. Info: MP2*LOIT E-Plan, EOP Use, NRC-2016

Justification

A - WRONG; The wind direction is not important to track during the initial cooldown to isolate the affected SG. The release of steam for this purpose is not defined as an unmonitored release.
Plausible; Examinee may recognize the changing wind direction will affect the PARs, which must be reported to the State DEP.

B - WRONG; Crossing the 200 °F subcooling line during the "blowdown" phase is not a classification condition.
Plausible; Examinee may believe violating the curve could imply a potential PTS situation, which would constitute degrading conditions.

C - CORRECT; 0% head level during a LOCA is a trigger for escalating the Event Classification and would not normally be expected on a "Small-Break" LOCA.

D - WRONG; During a loss of offsite power, it is not unexpected to see switchyard breakers opening and closing as CONVEX attempts to ascertain and solve the loss of power.
Plausible; Examinee may consider changes in the possible cause of the LOOP to be a required item to report.

References

EPA-REF02

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.4.39 **RO** 3.9 **SRO** 3.8 **CFR Link** (CFR: 41.10 / 45.11)

Knowledge of the RO's responsibilities in emergency plan implementation.

Question #: **74** | **Question ID:** **1000046** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

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

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

What is the reason for this action?

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

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

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

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

Question Misc. Info: MP2*LOIT, fire, 2559, MB-05666, NRC-2011 [067, AK3.04], NRC-2016

Justification

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

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

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

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

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

References

AOP 2559

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.4.27 **RO** 3.4 **SRO** 3.9 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of "fire in the plant" procedure.

Question #: **75** | **Question ID:** **156617** | ☑ **RO** | ☐ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The plant was tripped when the main condenser boot seal ruptured, causing a loss of condenser vacuum.

Other than the plant status described below, all other plant systems and components are operating normally.

The following conditions now exist:

- All actions of EOP 2525 have been completed.
- The crew has transitioned to EOP 2537, Loss of All Feedwater.
- Aux. Feedwater is NOT available, estimated time to restore is about 45 minutes.
- #1 S/G level = 115" and lowering, will reach 70" in 25 minutes.
- #2 S/G level 70" and lowering, will reach 32" in 10 minutes.
- Condensate pump feed to the S/Gs will be available in 15 minutes.

Which of the following actions is required Per EOP 2537, Loss of All Feedwater, assuming all given estimated times hold true?

☐ **A** Immediately transition to EOP 2540, Functional Recovery, then initiate Once Through Cooling if #1 S/G reaches 70" prior to feed flow being restored.

☑ **B** Immediately initiate Once Through Cooling and transition to EOP 2540, Functional Recovery, then continue efforts to restore feed to either of the S/Gs.

☐ **C** If #2 S/G approaches 32" before feed flow has been restored, initiate Once Through Cooling, and transition to EOP 2540, Functional Recovery, then continue efforts to restore feed to either S/G.

☐ **D** Immediately initiate Once Through Cooling and continue efforts to restore feed to either of the S/Gs, then if #1 S/G reaches 70" prior to restoration of feed flow, transition to EOP 2540, Functional Recovery.

Question Misc. Info: MP2*LOIT [000 537-01-B] 2537, EOP

Justification

A - WRONG; Because Once Through Cooling must be initiated if either S/G level reaches 70".
Plausible; Examinee may remember the 70" level requirement for OTC initiation and the need to transition to EOP 2540, but note a source of feed water will be available before both S/Gs are less than 70".

B - CORRECT; EOP 2537 requires Once Through Cooling to be initiated if either S/G level reaches 70", if main or auxiliary feedwater has not been restored. Because a S/G is about to drop below 70", Once Through Cooling must be immediately initiated.

C - WRONG; With a S/G dropping below 70" Once Through Cooling must be immediately initiated. It is required to be fully implemented before a S/G reaches 32".
Plausible; Examinee may remember the 32" requirement for OTC initiation but believe it is the level where initiation actions must be started.

D - WRONG; A LOAF combined with the manually triggered LOCA (OTC initiated) is considered two events, which requires immediate transition to EOP 2540 regardless of existing SG level.
Plausible; Examinee may recognize the need for implementation of OTC but not the need for the Functional Recovery EOP due to one S/G not meeting the OTC trigger value.

References

EOP 2537

Comments and Question Modification History

Removed second sentence of the first parapgraph.

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

Generic K/A Selected

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

Number 2.4.1 **RO** 4.6 **SRO** 4.8 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of EOP entry conditions and immediate action steps

Question #: **76** | **Question ID:** 9000003 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☑ Lower Order?

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

While operating at 100% power, the RCP A UPPER SEAL PRES HI annunciator alarms.

While referring to the appropriate Annunciator Response Procedure, the RCP A BLEED-OFF FLOW HI annunciator alarms.

Within a minute, the RCP A BLEED-OFF FLOW HI annunciator clears and the RCP A BLEED-OFF FLOW LO annunciator alarms and remains lit.

Numerous annunciators associated with "A" RCP seals also alarm, indicative of multiple RCP seal failures.

Which of the following describes the reason for this sequence of annunciators and the direction that must be given?

- ☐ **A** Bleedoff relief valve has failed open. Direct a Reactor and Turbine trip, secure the "A" RCP and go to EOP 2525, “Standard Post Trip Actions”.
- ☐ **B** Middle seal has failed. Evaluate the condition of the other seals per OP 2301C, “Reactor Coolant Pump Operation”, and ensure no other degradation or failures.
- ☐ **C** Vapor seal has failed. Perform AOP 2575, “Rapid Down power”, Reduce Reactor power to remove the Unit from Service and secure “A” RCP.
- ☑ **D** Bleedoff excess flow check valve has seated. Direct a Reactor and Turbine trip, secure the "A" RCP and go to EOP 2525, “Standard Post Trip Actions”.

Question Misc. Info: MP2*LOIT 2301C, RCP, RCS, MB-01952, NRC-2009, NRC-2016

Justification

SRO Justification: Questions meets 10CFR55.43.5, Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency conditions. SRO is responsible for evaluating the given conditions and deciding the appropriate course of action to take when multiple conflicting alarms are received.

A - WRONG; Opening of the RCP Bleedoff Relief Valve would likely result in a high Bleedoff flow on **all 4** RCPs and NOT a low Bleedoff flow annunciator. Additionally, "A" RCP Seal pressures are dependent on seal conditions and not necessarily the flow path of Bleedoff. Plausible because the examinee may be confused on how the Bleedoff Relief Valve would impact an individual RCP Bleedoff flow and pressure. The examinee may also be confused as to whether each RCP had a bleedoff relief valve.

B - WRONG; The individual indications provided could be a result of a failure of the "A" RCP Middle Seal; however, the indications together would indicate a loss of Bleedoff flow through the "A" RCP seals requiring the RCP to be stopped. If a Middle Seal had failed, then the action is correct. If Bleedoff Pressure Controller, PIC-215, had malfunctioned, then the action would be appropriate. Plausible because the Annunciator Response Procedures for the Upper Seal High Pressure and Bleedoff High Flow annunciators state that these alarms may be indicative of a failed Middle Seal.

C - WRONG; Failed RCP Vapor Seal will also cause a RCP A BLEED-OFF FLOW LO annunciator alarm but not a RCP A BLEED-OFF FLOW HI.
Plausible Failed RCP Vapor Seal will also cause a RCP A BLEED-OFF FLOW LO annunciator alarm when all of bleed off flow is going into Containment, in which case AOP 2575 would be a procedure to distract the Operator thinking the correct action would be to lower Reactor Power prior to tripping to minimize the perturbation on the RCS parameters to prevent exacerbating the leak.

D - CORRECT; The RCP A UPPER SEAL PRES HI annunciator is indicative of a failure of the "A" RCP Middle or Lower Seal (or a combination of both). This resulted the a high Bleedoff flow through the "A" RCP Seals resulting in a RCP A BLEED-OFF FLOW HI annunciator. At 10 gpm, the Excess Flow Check Valve will close causing the RCP A BLEED-OFF FLOW HI annunciator to clear and the RCP A BLEED-OFF FLOW LO to annunciate ((0.75 gpm). At this point, the RCP Seal package has NO cooling flow and the RCP must be tripped. Procedurally, the reactor and turbine are tripped prior to tripping the affected RCP.

References

ARP 2590B-069

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Question #: **76** | **Question ID: 9000003** | ☐ **RO** | ☑ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

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

Number 2.1.23 **RO** 4.3 **SRO** 4.4 **CFR Link** (CFR: 41.10 / 43.5 / 45.2 / 45.6)

Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Question #: **77** | **Question ID: 85552** | ☐ **RO** | ☑ **SRO** | ☐ Student Handout? | ☑ Lower Order?
Rev. 1 | ☑ Selected for Exam | **Origin: Bank** | ☐ Past NRC Exam?

AOP 2512 “Loss of All Charging” was entered per ARP “CHARGING FLOW LO” C02/3 A-15 AND due to a lowering PZR Level a Manual Plant Trip was performed. ALL actions of EOP 2525 “Standard Post Trip Actions” were performed and the Crew transitioned to EOP 2526 “Reactor Trip Recovery”

What further actions would be required to restore PZR Level?

☐ **A** Exit EOP 2526 "Reactor Trip Recovery" Go To OP 2207 "Plant Cooldown" depressurize and cooldown, start a HPSI pump to restore PZR Level.

☑ **B** Continue EOP 2526 "Reactor Trip Recovery" and Refer To AOP 2512 "Loss of All Charging", cooldown, depressurize, and start a HPSI pump to restore PZR Level.

☐ **C** Complete EOP 2526 "Reactor Trip Recovery" maintain the plant in MODE 3,Go To OP 2272C, "Plant Operation in MODE 3 Prior to Reactor Startup."

☐ **D** Continue EOP 2526 "Reactor Trip Recovery" and Refer To EOP 2541 appendix 8, "Plant Cooldown", cooldown, depressurize, and start a HPSI pump to restore PZR Level.

Question Misc. Info: AOP 2512, Loss of All Charging, CVCS

Justification

SRO Justification: Questions meets 10CFR55.43.5, Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency conditions.

AOP 2512 Discussion Section 1.2 last paragraph

A-WRONG; AOP 2512 is still required to be performed in parallel with the EOP 2526 Reactor Trip Recovery.
Plausible; Operator may think that AOP 2512 no longer applies following the Reactor trip, and a transition out of the EOP network is required to perform a Plant Cooldown.

B-CORRECT; With no Charging flow, the RX will be tripped without much delay and Operators will enter EOP 2525, “Standard Post Trip Action”. Once EOP 2525 and the Diagnostic Flow Chart are completed, if no other Optimal or Functional Recovery procedures are required, EOP 2526, “Reactor Trip Recovery” is entered and this procedure will be performed in parallel until Charging is restored or HPSI injection flow established. AOP 2512 paragraph 1.2

C-WRONG; Does not provide actions to restore PZR level.
Plausible; Operator may think that if a Reactor trip was done early in the AOP then PZR level would not go below requires actions and staying in mode 3 will allow the recovery of a Charging pump then a Plant Cooldown is not required.

D-WRONG; EOP 2526 does not reference EOP 2541 appendix 8 to cooldown the Plant.
Plausible; The Operator may think that it is required to stay in the EOP procedure network to restore PZR level therefore requiring the use of EOP 2541 Appendices.

References

AOP 2512

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.4.8 **RO** 3.8 **SRO** 4.5 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)
Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Question #: **78** | **Question ID:** **1653546** | ☐ **RO** | ☑ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

While conducting a plant cooldown the following plant conditions exist:

- RCS Temperature 235°F
- RCS pressure 245 psia
- 'A' LPSI Pump supplying Shutdown Cooling

If a Loss of Primary Coolant Accident occurs inside containment, which of the following actions are required?

☐ **A** Manually initiate Safety Injection flow by pressing the SIAS button on C-01. Go To EOP 2532, "Loss of Primary Coolant."

☐ **B** Manually start all available charging pumps and HPSI pumps, as necessary. Go To EOP 2532, "Loss of Primary Coolant."

☐ **C** Manually initiate Safety Injection flow by pressing the SIAS button on C-01. Go To AOP 2568A "RCS Leak, MODE 4, 5, 6, and Defueled.

☑ **D** Manually start all available charging pumps and HPSI pumps, as necessary. Go To AOP 2568A "RCS Leak, MODE 4, 5, 6, and Defueled.

Question Misc. Info: MP2*LORT 2207, HPSI, LOCA, SDC, RHR, NRC-2016

Justification

SRO Justification: Questions meets 10CFR55.43.5, because it requires the SRO to understand how the given RCS temperature places the plant in mode where the EOP 2532, LOCA , does not apply. Also, it requires knowledge of how the abnormal shutdown plant configuration of the applicable systems would necessitate the mitigating actions contained in the Plant Cooldown procedure and the RCS Leak AOP 2568A.

A - WRONG; At this point in Cooldown initiation of Safety Injection must be done manually, due to various unusual system alignments. Plausible; Examinee may remember at least one HPSI pump and two charging pumps are available for use and SIAS being blocked would require manual actuation, but believe that EOP 2532 would cover both conditions.

B - WRONG; EOP 2532 does not contain the proper guidance for mitigating this event due to the unusual system alignments that are required by Tech. Specs. and OP 2207.
Plausible; Examinee may feel that because the plant is not yet in Cold Shutdown, the LOCA EOP for "hot" Modes must be utilized.

C - WRONG; SIAS is inhibited and must be initiated manually by shifting valve positions and starting pumps.
Plausible; Examinee may believe that because many SI pumps are not actively disabled (and required to be available), that manual initiation of SIAS will work, as the manual button will override the "SIAS Block" presently active.

D - CORRECT; OP 2207, manually starts all available charging and HPSI pumps and transitions to AOP 2568A for continued actions to mitigate the RCS loss due to the complexities of various system alignments under these plant conditions.

References

AOP 2568A

Comments and Question Modification History

Second bullet replaced °F with psia added "A" after 2568

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

Generic K/A Selected

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

Number 2.4.6 **RO** 3.7 **SRO** 4.7 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of EOP mitigation strategies.

Question #: **79** | **Question ID:** 2016017 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

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

Which of the following sequence of events, occurring on a plant trip, would require reporting as an Emergency Action Level of Alert or above?

☑ **A** Main turbine trip on EHC system failure, "DIVERSE RX TRIP ACTUATED" and “PZR HI PRES TRIP” (4) alarms on C04, all CEAs inserting, manual reactor trip buttons pressed, then "REACTOR TRIP" alarm comes in and all Trip Circuit Breakers automatically open.

☐ **B** Inadvertent MSI triggers, "REACTOR TRIP", "DIVERSE RX TRIP ACTUATED" and “PZR HI PRES TRIP” (4) alarms on C04, all Trip Circuit Breakers opened, all CEAs inserting, manual reactor trip buttons pressed, ADV manually adjusted, then all MSSVs go closed.

☐ **C** Feedwater leak to #2 S/G inside CTMT, manual reactor trip buttons pressed, "REACTOR TRIP" alarm and all Trip Circuit Breakers open, CTMT pressure 5 psig and rising, MSI manually actuated on both facilities, then both Main Feed Pumps automatically trip.

☐ **D** High vibration on main turbine, manual reactor trip buttons pressed, "REACTOR TRIP" alarm on C-04 and all Trip Circuit Breakers open, all CEAs inserting, turbine fails to trip, MSIVs fail to close from C05, then both MSIVs closed from Bottle-up Panels (C-70A/B).

Question Misc. Info: MP2*LOIT CED-01-C, CEDS, ATWS, DSS, Classify, EAL, SRO, NRC-2016

Justification

SRO Only Justification; Emergency Action Level determination, and the specific definition of what constitutes an ATWS, is an SRO level knowledge and ability.

A - CORRECT; Even though the reactor was successfully tripped without the need for operator action, the choice states that TCBs don't open until manual trip buttons are pressed. This means the CEAs are inserting because of the automatic actions of the ATWS Mitigation Circuit (DSS) and NOT the auto actions of RPS. An ATWS is defined as a failure of RPS to trip the reactor, **not** a failure of all automatic trips of the reactor, because even though the DSS is required by law, no credit is given in the license (Tech. Specs.) for its operation. Therefore, this constitutes an "AUTOMATIC REACTOR TRIP FAILURE" under the Alert classification level.

B - WRONG; MSSVs remaining open post-trip, requiring operator action to lower SG pressure to close, does not constitute an Excess Steam Demand, provided the valves do go closed when action is taken at this time.
Plausible; Examinee may consider MSSVs failure to close requiring operator action as initially being an uncontrolled steam release outside of CTMT, which would classify as an Alert. (This was a past event on Millstone Unit 3).

C - WRONG; Failure of ESAS is not classified as an Alert or above, but does require eventual NRC notification.
Plausible; Examinee may believe failure of the Emergency Safety Actuation System to be a major event worthy of classification, especially where the reactor did automatically trip in the other choices.

D - WRONG; Failure of the turbine to trip automatically would not require an Alert classification as this is not an ATWS.
Plausible; Examinee may consider failure of the turbine to trip requiring action be taken outside of the control room to be a classifiable event.

References

EAL Tables required. EP-MP-26-EPI-FAP06-002

Comments and Question Modification History

Revised based on conflict with question #9 and attempt to reduce words. - RLC

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

Number EA2.02 **RO** 4.2 **SRO** 4.4 **CFR Link** (CFR 43.5 / 45.13)

Ability to determine or interpret the following as they apply to a ATWS: Reactor trip alarm

Question #: **80** | **Question ID:** **1100045** | ☐ **RO** | ☑ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

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

Electrical Maintenance has asked to take the turbine battery out of service for a couple hours to perform testing.

Which of the following describes the Technical Specifications concern if Electrical Maintenance were allowed to proceed?

☐ **A** On a loss of Inverter 1 or 2, concurrent with a loss of offsite power and failure of the opposite facility EDG, power from the Turbine Battery is required to ensure an AFAS can actuate the applicable components to feed the S/Gs.

☑ **B** On a loss of a Vital DC Bus concurrent with an Excess Steam Demand Event in containment, power from the Turbine Battery is required to ensure an MSI actuation can isolate Main Feed Water flow to the affected S/G.

☐ **C** On a loss of a Vital DC Bus, concurrent with a loss of offsite power, the Turbine Battery is required to ensure the actuation of both facilities of Emergency Safeguard Equipment during a design basis LOCA to prevent fuel melt.

☐ **D** On a loss of Inverter 1 or 2 concurrent with an Appendix "R" fire that requires control room evacuation, power from the Turbine Battery is required to ensure pressurizer level and RCS Inventory indication is still available.

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

Justification

SRO Only Justification; Requires knowledge of the Tech. Spec. Bases for the Turbine Battery and how it relates to detailed knowledge of the power supplies to various control systems.

A - WRONG; This is NOT the bases in Tech Specs for the Turbine Battery.
Plausible: The loss of VA-10 or VA-20 will prevent AFAS from actuating automatically if the opposite Vital AC bus is also lost. Because the Turbine Battery will supply VA-10 and VA-20 through INV-5 and INV-6 in this case, the examinee may believe it to be the basis from Tech Specs.

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

C - WRONG; Although true, this is NOT the bases in Tech Specs.
Plausible: The loss of a Vital DC Bus will cause a loss of normal power to VA-10 or VA-20 which, combined with a loss of Inverter 5 and 6 (powered from the Turbine Battery) as the backup power supply, would de-energize VA-10 or VA-20. This would prevent the actuation of the applicable facility of ESAS equipment. With the complete loss of one entire facility of safety components to actuate, the failure of any other component on the other facility would prevent the designed mitigating actions. The examinee should recognize this as a true statement and may believe that this is the basis for the Turbine battery.

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

References

U2-14-OPS-BAP05 3/4.8 Bases

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.2.36 **RO** 3.1 **SRO** 4.2 **CFR Link** (CFR: 41.10 / 43.2 / 45.13)

Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Question #: **81** | ***Question ID:*** ***2016019*** | ☐ ***RO*** | ☑ ***SRO*** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The plant tripped from 100% power due to a Large Break LOCA, with the following events and conditions:

- "D" CAR Fan tripped on overload.
- SIAS, CIAS, EBFAS, MSI and CSAS fully actuated for Facility 2.
- CTMT pressure is 24 psig and slowly rising.
- ALL other equipment is operating as designed.

The crew has completed all actions of EOP 2525, Standard Post Trip Actions and has transitioned to EOP 2532, Loss Of Coolant Accident, when the following occurs:

- RSST is lost due to an internal fault.
- "A" EDG frequency is indicating 60 hz, MWs are indicating zero (0).
- 24E is tied to 24C, and both are de-energized.
- 24D is energized on the "B" EDG.

Which one of the following statements describes the course of action the US must take?

☐ **A** Immediately attempt to energize Facility 1 Vital AC using EOP 2540F, CTMT Temperature and Pressure Control, then restore Facility 1 CTMT Spray to operation.

☐ **B** Immediately attempt to energize Facility 1 Vital AC using EOP 2540F, CTMT Temperature and Pressure Control, then restore Facility 1 CAR Fans to operation.

☑ **C** Immediately attempt to energize Facility 1 Vital AC using EOP 2541, Standard Appendices, then restore Facility 1 CTMT Spray to operation using EOP 2532, LOCA.

☐ **D** Immediately attempt to energize Facility 1 Vital AC using EOP 2541, Standard Appendices, then restore Facility 1 CAR Fans to operation using EOP 2532, LOCA.

Question Misc. Info: MP2*LOIT, LOCA, EOP 2532, RBCCW, NRC-2011, 55.43(b)(5), NRC-2016

Justification

SRO Justification: Questions meets 10CFR55.43.5, Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency conditions.

A - WRONG: EOP usage guidelines do not allow direct transition to a specific Functional Recovery Safety Function procedure.
Plausible: This action could possibly succeed, if it were allowed.

B - WRONG: EOP usage guidelines do not allow direct transition to a specific Functional Recovery Safety Function procedure and the Fac. 1 CAR fans cannot be used due to CTMT pressure preventing the restoration of Fac. 1 RBCCW.
Plausible: This action would succeed, if it were allowed.

C - CORRECT: The "A" EDG did not immediately reenergize the bus, therefore, the high CTMT pressure will prevent restoration of Fac. 1 RBCCW. This makes the Fac. 1 CAR Coolers unavailable for use, leaving only one CAR Cooler and the "B" Containment Spray Pump for Containment temperature and pressure control. Action must be taken to restore Additional Containment cooling. Restoration of power to Bus 24C will allow the "A" Containment Spray Pump to be placed in service and preserve the Containment Temperature and Pressure Control Safety Function. Procedure usage guidelines allow the use of Standard Appendices to restore power to needed equipment and recover a Safety Function, without the need to immediately transition to the Functional Recovery procedure.

D - WRONG: RBCCW can not be restored on Facility 1 due to CTMT pressure being >20 psig. Therefore the Facility 1 CAR Fans cannot be recovered.
Plausible: This action would work if it were not for the water hammer concern in the CAR Coolers.

References

EOP 2532-001

NO Comments or Question Modification History at this time.

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

Number AA2.21 **RO** 3.6 **SRO** 3.8 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Loss of Offsite Power: ED/G frequency and voltage indicators

Question #: **82** **Question ID:** **8000032** ☐ **RO** ☑ **SRO** ☐ **Student Handout?** ☐ **Lower Order?**

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

The crew has just opened up the TCBs to shut down the reactor for a refueling outage, when the following conditions are noted:

- Pressurizer pressure = ~2248 psia and slowly lowering
- Pressurizer level = ~ 39% and slowly lowering
- Letdown flow = ~ 40 gpm and slowly lowering
- Charging flow = 88 gpm and stable
- "C" charging pump is running in "lead"
- "A" charging pump is running in "manual"
- "B" charging pump is in "Pull-To-Lock".
- The STA has calculated an RCS leak rate of ~ 45 gpm.
- Process Radiation Monitor Hi-Hi/Fail annunciator alarmed
- S/G Blowdown automatically isolated

Based on the existing conditions, which of the following procedures is the US required to use to mitigate the event and continue the plant cooldown?

☑ **A** AOP 2569, Steam Generator Tube Leak, to isolate the affected Steam Generator. Then, complete the plant cooldown to Mode 5 using OP 2207, Plant Cooldown.

☐ **B** EOP 2534, Steam Generator Tube Rupture, to isolate the affected Steam Generator. Then, perform a plant cooldown to Mode 5 using EOP 2534, Steam Generator Tube Rupture.

☐ **C** AOP 2569, Steam Generator Tube Leak, to isolate the affected Steam Generator. Then, complete the plant cooldown to Mode 5 using AOP 2569, Steam Generator Tube Leak.

☐ **D** EOP 2534, Steam Generator Tube Rupture, to isolate the affected Steam Generator. Then, perform a plant cooldown to Mode 5 using EOP 2541, Appendix 12, SGTR Response.

Question Misc. Info: MP2*LOIT, SGTL, AOP 2569, OP 2205, NRC-2016

Justification

SRO Justification: Questions meets 10CFR55.43.5, Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency conditions.
A - 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 - WRONG; It would be inappropriate to enter EOP 2534 to isolate the affected S/G because the conditions are indicative of only a tube leak at this time, NOT a tube rupture. Even though the calculated leak rate is greater than the Charging/Letdown flow "mismatch", pressurizer level has NOT lowered enough to reduce letdown flow to the minimum value (28 gpm). With minimum letdown flow, the available charging pumps have the capacity to stabilize pressurizer level.
Plausible; Examinee may consider RCS input/output mismatch an indication of leakage exceeding charging capability, which would require the use of EO 2534.
C - WRONG; AOP 2569, Steam Generator Tube Leak, contains the guidance to start the cooldown, but lacks the guidance to complete the cooldown.
Plausible; Examinee may conclude that the AOP completes the cooldown based on the specific guidance given for tube leak mitigation being very close to that given in EOP 2534 for a SGTR (which does include actions for cooldown to Mode 5).

D - WRONG; EOP 2534 provides the guidance to isolate the most affected S/G; however, there is NO procedural guidance for the transition between EOP 2534 and EOP 2541, Appendix 12. Appendix 12 does NOT provide the guidance for performing a cooldown. It only provides guidance to isolate the affected S/G in the Functional procedures.
Plausible; Examinee may recognize the leak rate as entry conditions for EOP 2534 and, therefore, use of EOP 2541 as the cooldown method.

References

AOP 2569

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 037 Steam Generator (S/G) Tube Leak

Number AA2.04 **RO** 3.4 **SRO** 3.7 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: Comparison of RCS fluid inputs and outputs, to detect leaks

Question #: **83** | **Question ID: 2016042** | ☐ **RO** | ☑ **SRO** | ☐ Student Handout? | ☐ Lower Order?

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

A plant startup is in progress, power is currently 2% and rising with “A” SGFP in service.

A Steam Generator Tube Rupture occurs on the #2 S/G.

The Crew has transitioned to EOP 2534 “Steam Generator Tube Rupture” and commenced actions to mitigate the event.

Prior to isolating the Steam Generator:

1. What is the current status of a “Radiological Release” on the “Incident Report Form?”
2. How would the change in mitigating strategies, due to a loss of Main Condenser vacuum, affect the “Radiological Release?”

☐ **A** “Release in progress due to the event” which would change to “No radiological release due to the event”

☐ **B** “No radiological release due to the event” which would remain “No radiological release due to the event”

☐ **C** “Release in progress due to the event” which would remain “Release in progress due to the event”

☑ **D** “No radiological release due to the event” which would change to “Release in progress due to the event”

Question Misc. Info: MP2*LOIT Mode 2, SGTR, Mitigate, Rad Release, NRC-2016

Justification

SRO Justification: Questions meets 10CFR55.43(5), Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency conditions by understanding the requirements for reporting accident conditions to the State.

"A" WRONG; Currently there is no release in progress due to the steam release path is to the Main Condenser and incorrect for No release path because loss of the main condenser would require using the Atmospheric Dump Valves.
Plausible; Examinee may think the ADVs and Steam Dumps will quick open on the Reactor Trip but in fact does not at this low power level and may confuse Classification of a barrier with State Incident Report Form for Release which would miss interpret a cooldown using the ADVs as not a Prolonged Release.

"B" WRONG; Wrong currently there is no release in progress due to the steam release path is to the Main Condenser and incorrect for No release path because loss of the main condenser would require using the Atmospheric Dump Valves.
Plausible; No release is incorrect and Examinee may confuse Classification of a barrier with State Incident Report Form for Release which would miss interpret a cooldown using the ADVs as not a Prolonged Release.

"C" WRONG; Wrong currently there is no release in progress due to the steam release path is to the Main Condenser and correct for a release path because loss of the main condenser would require using the Atmospheric Dump Valves.
Plausible; Examinee may think the ADVs and Steam Dumps will quick open on the Reactor Trip but in fact does not at this low power level.

"D" CORRECT; Currently there is no release in progress due to the steam release path is to the Main Condenser and correct for a release path because loss of the main condenser would require using the Atmospheric Dump Valves.

References

EPI-FAP06

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 051 Loss of Condenser Vacuum

Generic K/A Selected

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

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

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

Question #: **84** **Question ID:** **2016021** ☐ **RO** ☑ **SRO** ☐ **Student Handout?** ☑ **Lower Order?**

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

A fire has occurred in the 25'6" Auxiliary Building Cable Vault. Fire suppression activities appear to be causing uncontrolled component operations in the plant. The SM orders a Control Room evacuation and control of the plant be restored using AOP 2559, "Fire" and AOP 2579A, "Fire Procedure for Hot Standby Appendix R Fire Area R-1".

Which of the following plant components would be rendered not OPERABLE or capable of performing its design function, once control of the plant is regained per AOP 2579A?

☑ **A** Power Operated Relief Valves (PORVs).

☐ **B** Main Steam Isolation Valves (MSIVs).

☐ **C** Atmospheric Dump Valves (ADVs).

☐ **D** Auxiliary Feed Regulating Valves (AFRVs).

Question Misc. Info: MP2*LOIT Fire, App. "R", AOP, 2559, TS, TSAS, Bases, SRO, NRC-2016

Justification

SRO Justification; 10CFR55.43(1) SRO needs to understand the Conditions and Limitation of the facility license to determine OPERABILITY for components knowing the required Tech. Spec.

A - CORRECT; The AOPs require the switches on the Bottle-up Panels be placed in isolate for the PORVs, MSIVs and ADVs. This will fail closed both facilities of all three of these components. It also requires all control power except VA-20 be deenergized, failing open the Fac. 1 AFRV. However, IAW Tech. Spec. Bases, the PORVs are suppose to be available for RCS pressure control if needed.

B - WRONG; The MSIVs are in their accident position when closed.
Plausible; Procedurally required when evacuating the control room for a fire, the operators are required to place all isolation switches in the Bottle Up Panel to “ISOLATE” therefore for both MSIV the, Examinee may determine this action INOPs the valves.

C - WRONG; The ADVs are not required to be operated remotely from the control room to perform their design function.
Plausible; Procedurally required when evacuating the control room for a fire, the operators are required to place all isolation switches in the Bottle Up Panel to “ISOLATE” , which isolates all control signals to the "A" ADV and isolates all control signals to the "B" ADV except those from C-10. Therefore, the Examinee may determine that this action INOPs both ADVs.

D - WRONG; The AFRVs are in their accident position when failed open.
Plausible; Procedurally required to evacuate to Fire Shutdown Panel C-10 the Examinee may think that since there are no controls for Facility 1 Aux Feed Reg Valve that this would make the Valve not OPERABLE.

References

AOP 2559

NO Comments or Question Modification History at this time.

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

Number AA2.14 **RO** 3.2 **SRO** 4.3 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Plant Fire on Site: Equipment that will be affected by fire suppression activities in each zone

Question #: **85** **Question ID: 2016023** ☐ RO ☑ SRO ☐ Student Handout? ☑ Lower Order?
Rev. 0 ☑ Selected for Exam **Origin: New** ☐ Past NRC Exam?

The plant is at 90% power with a shutdown in progress using AOP 2575, "Rapid Downpower", due to an RCS leak from an unknown location.

Which of the following RCS leak indications would require the US transition to a different procedure to shut down the plant?

☐ **A** Alarm on RBCCW radiation monitor and Chemistry reports unable to draw an RCS sample due to high temperature.

☐ **B** Alarm on Pressurizer Level Low and Main Turbine controls are in "Hold" mode on the EHC Insert panel.

☐ **C** Alarm on "RX VESSEL HEAD SEAL LEAK-OFF TEMP HI" with leak-off temperature at 205°F prior to closing the RX flange valve.

☑ **D** Alarm on "Main Steam Line Radiation Monitor" with two charging pumps running and pressurizer level continuing to lower.

Question Misc. Info: MP2*LOIT RCS Leak, CTMT, 120VAC/125VDC, NRC-2016

Justification

SRO Justification; 10CFR55.43(5) SRO needs to assess the facility conditions and selection of appropriate procedures during normal, abnormal and emergency conditions. During a rapid downpower the SRO assess plant conditions to understand that a S/G tube rupture has occurred requiring a reactor trip.

A - WRONG; Although this is an indication of an RCS leak outside of CTMT, it does not warrant a Rx trip in an of itself, so long as PZR level can be maintained with the available charging pumps.
Plausible; Examinee may recognize that this indicates the RCS leak is also a break in the CTMT barrier (2/3 barriers) and feel a plant trip in therefore required.

B - WRONG; With the turbine controls in 'hold' mode, the main turbine would not be lowering steam demand as the reactor is shutting down, resulting in an RCS cooldown and shrinkage. Therefore, even though this alarm indicates the PZR level control system is not successfully restoring PZR level, it would not necessarily indicate the RCS leakage is in excess of charging capacity, requiring a plant trip.
Plausible; Examinee may recognize the alarm for what it is and believe the turbine controls indicate conditions are stable (which normally would be correct).

C - WRONG; This indicates the inner o-ring is failing, but not the outer o-ring. Therefore a trip is not required.
Plausible; Examinee may see indication as degrading plant conditions due to head o-ring failure, warranting a plant trip.

D - CORRECT; MSL Rad. Monitor alarm indicates the RCS leak is due to a S/G tube leak. Although not all three charging pumps are running trying to restore PZR level, administrative requirements prevent the use of the third charging pump during AOP-2575. Therefore, the given charging pump status would meet the required validation criteria of the MSL rad monitor alarm and combined would indicate the leak is now a 'tube rupture', which requires a plant trip.

References

AOP 2569

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.4.45 **RO** 4.1 **SRO** 4.3 **CFR Link** (CFR: 41.10 / 43.5 / 45.3 / 45.12)

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

Question #: **86** | **Question ID: 2016031** | ☐ **RO** | ☑ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

Initial Conditions:

- Loss of Offsite Power
- RCS at NOP/NOT
- Three (3) CEAs are stuck fully withdrawn
- 24D is de-energized due to a bus fault
- "B" Charging pump is Out of Service for Maintenance

Crew has transitioned to EOP 2528 “Loss of Offsite Power"

Then the "A" charging pump discharge relief valve fails full open at this time.

Which of the following describes an action needed to satisfy the Reactivity Control Safety Function and the Procedural guidance for that action?

☐ **A** Transition to EOP 2540 “Functional Recovery” and initiate FSG-05 “Initial Assessment and FLEX Equipment Staging” to restore charging.

☐ **B** Remain in EOP 2528 “Loss of Offsite Power” and initiate a cooldown maintaining Reactor Power < 1x10-4 stable or dropping.

☑ **C** Transition to EOP 2540 “Functional Recovery” and depressurize the RCS, emergency borate using the HPSI pumps > 40 gpm.

☐ **D** Remain in EOP 2528 “Loss of Offsite Power” and depressurize the RCS, emergency borate using the HPSI pumps > 40 gpm.

Question Misc. Info: MP2*LOIT CVC-01-C,

Justification

SRO Justification: Questions meets 10CFR55.43.5, Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency conditions.

"A" WRONG; Transitioning to EOP 2540 is correct, implementation of the Flex procedure is only completed during a station black out.
Plausible; The examinee may assume that a loss of the grid allows the use of the FLEX procedures with no installed Charging pumps that are available allowing for the use of the new FLEX Equipment, until power to the vital 480 bus can be restored. No installed charging pumps available therefore a flex charging pump maybe a viable alternative.
"B" WRONG; < 1x10-4 stable or dropping is part of the Reactivity Safety Function which also requires all CEAs inserted or SD margin +350 ppm boron for each CEA not inserted for EOP 2528 therefore the US must re-diagnose using the Flow Chart which would require transition to EOP 2540.
Plausible; The examinee may know that 1x10-4 meets the safety function for another EOP therefore apply it for this condition only having to refer to the Emergency Boration appendix for further guidance on shutdown margin and a cooldown to reduce the impact on inventory reserves.
"C" CORRECT; When a safety function is not satisfied for the current EOP the US must re-diagnose. The loss of the running charging pump means conditions are not meeting the Reactivity Safety Function and the US must transition to EOP 2540 which would direct depressurizing the RCS by controlling RCS heat removal to meet Emergency Boration using the HPSI pumps.
"D" WRONG; < 1x10-4 stable or dropping meets the Reactivity Safety Function only when transitioning to EOP 2540.
Plausible; EOP 2528 allows for depressurizing the RCS the examinee will may believe that continuing and not transitioning will allow the same results by borating using HPSI but depressurization of the RCS without charging is not addressed in EOP 2528.

References

EOP 2541-APP01

NO Comments or Question Modification History at this time.

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/ 43/5 / 45/3 / 45/5)

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

Question #: **87** | **Question ID: 2016024** | ☐ **RO** | ☑ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**
Rev. 1 | ☑ **Selected for Exam** | **Origin: Mod** | ☐ **Past NRC Exam?**

A plant shutdown is in progress with reactor power at ~9% and dropping slowly.

RPS Channels "A" and "C" Linear Power Range Level 1 bistables have been reset (LEDs have gone out). However, Channels "B" and "D" Linear Power Range Level 1 bistables have failed in the "armed" state and will NOT reset (the LED remains lit and is not blinking). ALL other plant and RPS components are operating normally and are expected to continue functioning as designed.

Then, the board operators report the reactor shutdown has gotten to far ahead of the turbine resulting in an uncontrolled cooldown and suggest immediately shutting down the main turbine.

Which one of the following describes the effect of tripping the main turbine at this time and the action the US should direct based on these conditions?

☑ **A** The reactor will automatically trip.
Per OP 2205, "Plant Shutdown", precautions, immediately trip the reactor and the turbine and transition to EOP 2525, "Standard Post Trip Actions".

☐ **B** The reactor will automatically trip.
Refer to OP 2206, "Reactor Shutdown" and simultaneously insert all CEAs, trip the turbine and transition to EOP 2525, "Standard Post Trip Actions".

☐ **C** An uncontrolled positive reactivity addition will result due to feedwater control valve response.
Per OP 2205, "Plant Shutdown", precautions, immediately trip the reactor and the turbine and stabilize the plant using OP 2205, "Plant Shutdown".

☐ **D** An uncontrolled positive reactivity addition will result due to feedwater control valve response.
Refer to OP 2206, "Reactor Shutdown" and simultaneously insert all CEAs, trip the turbine and stabilize the plant using OP 2205, "Plant Shutdown".

Question Misc. Info: MP2*LOIT RPS-01-C, NRC, Reference: AOP 2575, step 4.38

Justification

SRO Only Justification: The question requires an in-depth knowledge of system interrelations at this point in a plant shutdown, as well as the administrative understanding of the defined reactor shutdown methods, in order to asses the plant conditions and select the appropriate procedure to transition to. Therefore, this is an SRO question because it meets 10 CFR 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

A - CORRECT; Based on guidance in OP 2206, "Reactor Shutdown", if a reactor trip is required due to plant conditions other than a normal shutdown, the crew is required to consider it a "reactor trip" (not a "shutdown"), and transition to EOP 2525, "Standard Post Trip Actions".

B - WRONG; The "simultaneous insertion of CEAs" is not applicable if the reactor is "tripped" in Mode 1.
Plausible; The "simultaneous insertion" of CEAs was defined to allow for the use of the reactor manual trip buttons to quickly shutdown the reactor when it had been manually driven to the low, unstable range of power. If power were about a decade or so lower, this option would be acceptable.

C - WRONG; Even though power is very low, a reator trip due to unstable plant conditions requires transition to EOP 2525.
Plausible; An uncontrolled cooldown could result if the turbine is tripped due to automatic feed control response and based on the given conditions, Plant Shutdown will be the probable procedure to transition to once EOPs are exited.

D - WRONG; Simultaneous insertion is not applicable for the given conditions.
Plausible; An uncontrolled cooldown could result and the given conditions would allow transition to OP 2205 once EOP exit criteria is met.

References

ARP 2590C-055, OP 2206

Comments and Question Modification History

Revised to rev. 1 based on validator feedback. - RLC

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

Number A2.01 **RO** 3.1 **SRO** 3.6 **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.5)

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

Question #: **88** | ***Question ID:*** ***2016026*** | ☐ ***RO*** | ☑ ***SRO*** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

The Plant is operating at 100% power.

"A" EDG Operability Test SP 2613A is in progress.

After the EDG is started the following alarms are received at C08:

- A-36 "DIESEL GEN 12U TROUBLE"
- D-30 "DIESEL GEN 12U DISABLED"

AND Locally at C-38:

- E-3 "ENGINE START FAILURE"
- C-7 "STARTING AIR PRESSURE LOW"

The report from the field was the Diesel rotated then stopped after about 10 seconds, and an abnormal air hissing sound that stopped at the same time the alarms came in.

After the US has evaluated one EDG NOT OPERABLE, which of the following actions are required as a consequence of this malfunction?

☑ **A** Refer to the ARP to restore Air Start pressure.
Determine no common mode failure on the "B" EDG.
Verify TDAFW is OPERABLE.
Restore to OPERABLE within 72 hours.

☐ **B** Refer to OP 2346A "A EDG" and air roll the "A" EDG.
PRESS "ALARM RESET" (engine skid) on the "A" EDG.
Ensure Station Black Out Diesel OPERABLE.
Restore to OPERABLE within 14 days.

☐ **C** Refer to the ARP to restore Air Start pressure.
Determine no common mode failure on the "B" EDG.
Ensure Station Black Out Diesel OPERABLE.
Restore to OPERABLE within 14 days.

☐ **D** Refer to OP 2346A "A EDG" and air roll the "A" EDG.
PRESS "ALARM RESET" (engine skid) on the "A" EDG.
Refer to SP 2346A to restart diesel per ARP 2591A.
Restore to OPERABLE within 72 hours.

Question Misc. Info: MP2*LOIT EDG, 2346A/B, NRC-2016

Justification

SRO Justification; 10CFR55.43(1)(5) SRO needs to understand the Conditions and Limitation of the facility license for Emergency Diesels, knowing the required Tech. Spec. action statement time limits beyond 1 hour and assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency conditions.

"A" CORRECT; The operator in the field would continue with the ARP locally and the US must perform the actions of the LCO checking for common mode failure or perform a surveillance run on the "B" EDG, verify the TDAFW pump OPERABLE and restore within 72 hours

"B" WRONG; Performing an Air roll of the EDG without determining the cause of the Start Failure may cause more damage to the EDG and per the ARP for START FAILURE an inspection for air leaks must be conducted, further more a 14 day action statement is only allowed for planned EDG maintenance.
Plausible; The Examinee may remember that a failed attempted start of the EDG requires in Air roll to prevent oil from the upper pistons leaking into the cylinder.

"C" WRONG; Ensuring the SBO is OPERABLE is only required for the 14 day action statement and is only allowed for planned EDG maintenance.
Plausible; Examinee may assume that ensuring the OPERABILITY of the SBO will allow them to extend the outage time for the EDG to 14 days.

"D" WRONG; Performing an Air roll of the EDG without determining the cause of the Start Failure may cause more

Question #: **88** **Question ID: 2016026** ☐ RO ☑ SRO ☐ Student Handout? ☐ Lower Order?

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

damage to the EDG and per the ARP for START FAILURE an inspection for air leaks must be conducted and pressing the Alarm Reset will allow the EDG to respond to an emergency signal.

Plausible; The actions described are for ENGINE START FAILURE alarm response which still must be accomplished but does not address the LCO

References

TS 3.8.1.1

NO Comments or Question Modification History at this time.

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

Number A2.01 **RO** 3.1* **SRO** 3.3 **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure modes of water, oil, and air valves

Question #: **89** | **Question ID:** **2016016** | ☐ **RO** | ☑ **SRO** | ☐ Student Handout? | ☑ Lower Order?

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

A hostile force gains access to the Protected Area and there are reports of explosions inside of the Unit 2 Intake Building.

Security Shift Operations Supervisor notifies the Control Room that a Station Blackout is imminent.

Unit Supervisor is required to notify the Security Shift Operations Supervisor and perform which of the following actions?

☑ **A** In C OP 200.2, “Security Event”, brief and dispatch a Plant Equipment Operator to align Fire Water to the selected Emergency Diesel Generator.

☐ **B** In C OP 200.2, “Security Event”, establish Search teams and dispatch to perform Attachment 2 "Unit 2 Search Checklist" starting with the AFW and Diesel areas.

☐ **C** Transition to AOP 2551, “Shutdown from Outside the Control Room”, and trip the Reactor and have all on-shift personnel report to C-21.

☐ **D** Transition to AOP 2579P, “Fire Procedure for Hot Standby Appendix R Fire Area R-16", have all on-shift personnel report to C-10.

Question Misc. Info: MP2*LORT*EPlan Security Event, NRC-2016

Justification

This is an SRO question because it meets 10 CFR 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

"A" CORRECT; C OP 200.2 was recently changed to include pre-planned routes are established to supply the Emergency Diesel Generator with Firewater in the advent of a hostile action on site ensuring a vital AC source, see attached.

"B" WRONG; Search teams are only required during a call in bomb threat.
Plausible; The Examinee will understand that the AFW and Diesel areas will be vital following an attack to the Intake and Transformer Yard with an Imminent Station blackout, and will want to ensure no threat to the vital equipment.

"C" WRONG; Evacuation of the Control Room is not required with an onsite actual threat.
Plausible; The Examinee may think that the Control Room is a targeted area which requires evacuation and the implementation of EOP 2551 Tripping the Reactor and required transition to C-21 that is the normal area for the Control Room staff to go when evacuating the CR.

"D" WRONG; Evacuation of the Control Room during a Hostile Event is not required per C OP 200.2.
Plausible; The Examinee may think that the explosions in the Intake requires implementation of AOP 2579P while that would be true if a Hostile Force was not attacking the Site, C OP 200.2 takes precedence during the Attack.

References

C OP 200.2, Step 4.2.2b

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.4.30 **RO** 2.7 **SRO** 4.1 **CFR Link** (CFR: 41.10 / 43.5 / 45.11)

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

Question #: **90** | **Question ID:** **2016035** | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

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

The Unit has just tripped due to a Statewide Blackout (the grid is lost), at the completion of EOP 2525 “Standard Post Trip Action” immediate actions, the following plant conditions exist:

- Bus 24C energized from the EDG
- Bus 24D de-energized EDG Emergency Tripped, No Service Water Pump
- Bus 24E is aligned to Bus 24D
- “A” AFW Pump P-9A Overload Tripped
- Instrument air is cross tied to Station Air
- All other equipment function as designed

With the following Main Board C06/07 Alarms in

- AA-1 “IAC F3E TROUBLE”
- AB-1 “T52E IA DRYER TROUBLE”

What direction should the Unit Supervisor provide to the Crew to optimize all Safety Functions?

☐ **A** Direct a PEO to manually cross-tie Station Air to Unit 3 per EOP 2540 Appendix 4A
Direct BOP to request Unit 3 provide power from the SBO

☑ **B** Direct the PEO to restart “E” IAC and Bypass the Dryer per ARP 2590E alarm response
Direct BOP to request Unit 3 provide power from the SBO

☐ **C** Direct a PEO to manually cross-tie Station Air to Unit 3 per EOP 2540 Appendix 4A
Direct BOP to request Unit 3 provide power from their Non Vital Bus (34A or 34B)

☐ **D** Direct the PEO to restart “E” IAC and Bypass the Dryer per ARP 2590E alarm response
Direct BOP to request Unit 3 provide power from their Non Vital Bus (34A or 34B)

Question Misc. Info:

Justification

SRO Justification; This is an SRO question because it meets 10 CFR 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

A - WRONG; With the Site experiencing a grid loss, Unit 3 will only have their Vital Buses energized, to be able to give Unit 2 Station Air U3 must energize their non-vital bus using the Station Blackout Diesel. The SBO can only be aligned to one of the following at a time U3 Bus 34A or 34B or U2 Bus 24E.
Plausible; If the Examinee does not understand that U3 bus 34A and 34B are dead due to offsite power loss and doesn’t understand the requirements to align Station Air from U3 they may assume that U3 can provide both.

B - CORRECT; Instrument Dyer Alarms, do not prevent air from supplying the IA header and the dryers are not a required component to supply air to the Header. Following a loss of power to the Air Compressor when the EDG re-energizes with no SIAS actuation, an Operator is required to restart the compressor locally. Energizing Bus 24E from the SBO will allow U2 to maintain defense in depth by providing swing components and allow for further flexibility by then energizing bus 24D

C - WRONG; Unit 3 cannot supply both air and power when on their Emergency Diesels, Bus 34A / 34B are both dead during a loss of offsite power and the SBO can only be aligned to one of the following at a time U3 Bus 34A or 34B or U2 Bus 24E.
Plausible; If the Examinee does not understand that U3 bus 34A or 34B is dead due to offsite power loss and doesn’t understand the requirements to align Station Air from U3 they may assume that U3 can provide both.

D - WRONG; Bus 34A / 34B are both dead during a loss of offsite power and only 2 of the following 4 breakers can be closed at a time; U3 Bus 34A or 34B or U2 Bus 24E or SBO output therefore power to 24E can only come from SBO diesel.
Plausible; U2 can credits U3 bus 34A / 34B as an alternate source for an Offsite line and performs a nightly surveillance, thereby the examinee may assume bus 34A / 34B as another source of power

References

ARP 2590E-002, OP 2332B, LP EOP-2530

NO Comments or Question Modification History at this time.

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

Number A2.01 **RO** 2.4 **SRO** 2.9 **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Ability to (a) predict the impacts of the following mal- functions or operations on the IAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Air dryer and filter malfunctions

Question #: **91** | **Question ID:** 2016037 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 0 | ☑ Selected for Exam | **Origin:** Bank | ☐ Past NRC Exam?

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

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

Which procedure and mitigating strategy will the operators implement next?

☐ **A** EOP 2532, Loss of Coolant Accident; Depressurize the RCS to mitigate the inventory loss.

☐ **B** EOP 2534, S/G Tube Rupture; Depressurize the RCS to mitigate the inventory loss.

☐ **C** EOP 2536, Excess Steam Demand; Slowly feed #2 SG only to mitigate the RCS cooldown.

☑ **D** EOP 2540, Functional Recovery; Slowly feed #2 SG only to mitigate the RCS cooldown.

Question Misc. Info: MP2 LOIT/LOUT, SRO, E25-01-C MB-2532, 10CFR43(b)(5), MB-05433, NRC-2002, NRC-2005, NRC-2016

Justification

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

A - WRONG: Multiple events (SGTR and ESDE) require entry into the functional recovery procedure.
Plausible: Lowering PZR level indicative of inventory loss. Depressurize the RCS to lower the leak rate.

B - WRONG: Multiple events (SGTR and ESDE) require entry into the functional recovery procedure.
Plausible: Pressurizer pressure dropping, no containment rad monitor alarms. Depressurize the RCS to lower the leak rate.

C - WRONG: Multiple events (SGTR and ESDE) require entry into the functional recovery procedure.
Plausible: #1 S/G level is dropping, subcooling is rising. Secure feeding #1 SG for the ESD to minimize RCS temperature drop.

D - CORRECT; Multiple events are in progress (SGTR and ESDE with failure of MSI), requiring entry into the functional recovery procedure. Control feed to the SGTR (and not the ESD) to mitigate the RCS cooldown.

References

EOP 2541-APP01

Comments and Question Modification History

Modified last two bullets to correct technical issue of MSL Rad Monitor rising post-trip (no N-16) and "no **other** rad monitors in alarm" - RLC

NRC K/A System/E/A **System** 002 Reactor Coolant System (RCS)

Number A2.02 **RO** 4.2 **SRO** 4.4 **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.5)

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

Question #: **92** | **Question ID: 2016029** | ☐ RO | ☑ SRO | ☐ Student Handout? | ☑ Lower Order?

Rev. 0 | ☑ Selected for Exam | **Origin: New** | ☐ 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 the effect on Group 7 rod withdrawal and requirements to meet Procedural and Technical Specifications, assuming all other plant conditions remain normal?

☐ **A** Record reed switch position indication and baseline data on local pulse counter for each CEA within 1 hour, or be in at least Hot Standby within the next 4 hours.

☑ **B** Record reed switch position indication and baseline data on local pulse counter for each CEA prior to movement and/or once a shift, or restore to OPERABLE within 24 hours.

☐ **C** Withdraw Group 7 to the UEL limit within 1 hour and record reed switch position indication and baseline data on local pulse counter for each CEA within the next 24 hours.

☐ **D** Withdraw Group 7 to $\geq$ 172 steps and perform a baseline on local pulse counter within 1 hour and baseline data on local pulse counter for each CEA within the next 24 hours.

Question Misc. Info: MP2*LOIT CED-01-C, CEDS, Position Indication, NRC-2016

Justification

SRO Justification; 10CFR55.43(1) SRO needs to understand the Conditions and Limitation of the facility license for Rod Control, knowing the required Tech. Spec. action statement time limits beyond 1 hour.

"A" WRONG; Baseline data on pulse counter is not required within 1 hour and describes TSAS 3.1.3.3.b.
Plausible; Surveillance requirements requires pulse counter baseline within 1 hour of rod motion > 10 steps

"B" CORRECT; Described actions from the AOP 2556 ("CEA Malfunctions"), OP 2302A ("Control Element Drive System") and TSAS 3.1.3.3.d ("Movable Control Assemblies - Position Indicator Channels") for a loss of ALL pulse counters.

"C" WRONG; This actions describes if a Reed Position indication were inoperable
Plausible; This describes the action required for a reed position indication failed with a CEA not in the Full in or Full out position

"D" WRONG; This action describes the requirement for meeting PDIL
Plausible; Examinee would assume with the loss of CEA position the requirement to ensure CEAs are above the PDIL for all power levels would be a priority.

References

TS 3.1.3.3, AOP2518

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.4.11 | **RO** 4.0 | **SRO** 4.2 | **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of abnormal condition procedures.

Question #: **93** | **Question ID: 2016036** | ☐ **RO** | ☑ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**
Rev. 0 | ☑ **Selected for Exam** | **Origin: New** | ☐ **Past NRC Exam?**

The plant is in MODE 6 with the following conditions:

- Fuel movement is in progress.
- The Personnel Airlock Doors are open
- The Equipment Hatch is removed.
- Containment Purge is in operation.

I&C working on ESAS inadvertently triggers a CIAS

What procedural guidance and action must be taken and why?

☑ **A** Refer to OP 2314B "CTMT and Enclosure Bldg Purge" secure CTMT Purge due to the loss of Main Exhaust Fans.

☐ **B** Refer to OP 2313A "CTMT Air Recirculation and Cooling System" over-ride and stop CAR fans that do not have RBCCW aligned.

☐ **C** Refer to OP 2310 "Shutdown Cooling System" and make manual adjustments to SDC temperature due to valve repositioning.

☐ **D** Refer to OP 2332A "Station Air" over-ride and open 2-SA-19 "CTMT Station Air Hdr. Isol." To maintain S/G Nozzle Dams.

Question Misc. Info: MP2*LOIT CTMT, Purge, CPS, SRO, NRC-2016

Justification

SRO Justification: IAW 10 CFR 55.43 (5), SRO required knowledge to evaluate the impact of a CIAS signal on the current plant conditions, based on (SRO knowledge of) the perquisites for Fuel movement in CTMT with the Equipment Hatch removed (i.e.; CTMT purge using Main Exhaust).

A - CORRECT; Inadvertent ESFAS AOP 2571 for CIAS causes all Main Exhaust Fans to trip, when Containment Purge in service it will require at the minimum 1 Main Exhaust Fan, therefore OP 2314B "CTMT and Enclosure Bldg Purge" secure purge operations.

B - WRONG; Inadvertent CIAS does not cause CAR fans to Auto start.
Plausible; If the Examinee thinks that a SIAS and CIAS are generated at the same time they may assume that the CAR fans auto start on a SIAS and that some coolers will not have RBCCW flow for the SDC configuration which they may interpret requiring manual over ride and securing fans.

C - WRONG; CIAS signal does not close or cause any valves in the SDC flow path to re-align or the RBCCW cooling flow path or the Service Water flow path.
Plausible; The Examinee may think that a system change to either SDC, RBCCW or Service Water would affect the RCS temperature requiring an adjustment to SDC system to counter act any system changes in the heat removal flow path.

D - WRONG; Although Station Air to CTMT isolates on a CIAS, the Steam Generator Nozzle Dams do not require any Air pressure to continue operations and will not fail.
Plausible; The Examinee may believe that the S/G nozzle dams require air to maintain its continuous function that would be incorrect in that it will continue to function without air. The system will alarm when air pressure does lower.

References

OP 2314A, AOP 2571

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 029 Containment Purge System (CPS)

Number A2.01 **RO** 2.9 **SRO** 3.6 **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 Containment Purge System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Maintenance or other activity taking place inside containment

Question #: **94** | **Question ID:** 5000026 | ☐ RO ☑ SRO | ☐ Student Handout? | ☑ Lower Order?
Rev. 1 | ☑ Selected for Exam | **Origin:** Bank | ☐ Past NRC Exam?

The plant is in MODE 6 with refueling operations in progress.

One Wide Range Excore Nuclear Instrument has failed; repairs are in progress.

I&C reports that two of the remaining wide range channels should be considered NOT OPERABLE due to inaccurate calibration.

The remaining channel was properly calibrated.

What impact does this have on fuel handling activities and why?

☐ **A** All fuel movement in containment and the spent fuel pool must be suspended due to inadequate remaining instrumentation for monitoring the state of the core.

☑ **B** All fuel movement in to and out of the reactor core must be suspended due to inadequate remaining instrumentation for monitoring the state of the core.

☐ **C** Fuel movement may continue since the operability of the remaining channel is adequate for monitoring the state of the core.

☐ **D** Fuel offload activities may proceed; fuel reload must be suspended due to inadequate remaining instrumentation for monitoring positive reactivity additions.

Question Misc. Info: MP2*LOIT NIS-01-C, CFR 55.43(6), NRC-2005 [K/A; 032, AK3.02], NRC-2016

Justification

SRO Justification; 10CFR55.43(1) SRO needs to understand the Conditions and Limitation of the facility license for Fuel Movement knowing the required Tech. Spec. actions.

CHOICE (A) - NO WRONG: CORE ALTERATIONS must be suspended without two operable channels, activities in the spent fuel pool are not CORE ALTERATIONS.
VALID DISTRACTOR: Plausible that all fuel handling would be stopped.

CHOICE (B) - YES CORE ALTERATIONS must be immediately suspended; fuel movement in the core is a subset.

CHOICE (C) - NO WRONG: Minimum channels operable requirement is TWO source range channels.
VALID DISTRACTOR: Plausible that one channel sufficient for core alterations.

CHOICE (D) - NO WRONG: All CORE ALTERATIONS must be suspended, not just those involving positive reactivity additions.
VALID DISTRACTOR: Plausible that only concerned about addition of positive from new fuel loading.

SRO Justification: The questions involves an in-depth knowledge of the applicable Tech. Spec. and Fuel Movement is an SRO only task.

References

OP 2209A

Comments and Question Modification History

Modifications to NRC-2005 question:
Changed "inoperable" to "NOT OPERABLE" in the last sentence before the question statement, to comply with correct word designation for Tech. Spec. "inoperability".
Deleted "Note" in Justification related to question use and 10CFR55.43(6) reference (info located in linked K/A).
Added SRO justification.

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

Generic K/A Selected

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

Number 2.1.36 **RO** 3.0 **SRO** 4.1 **CFR Link** (CFR: 41.10 / 43.6 / 45.7)

Knowledge of procedures and limitations involved in core alterations.

Question #: **95** **Question ID: 9000024** ☐ RO ☑ SRO ☐ Student Handout? ☑ Lower Order?
Rev. 2 ☑ Selected for Exam **Origin: Bank** ☐ Past NRC Exam?

The plant is operating at 100% power when ISO New England and CONVEX operators notify Millstone Station that a "Degraded Voltage" condition exists. Voltage on the 4.16 kV buses is presently 3,900 volts and the crew has entered AOP 2580, "Degraded Voltage".

Based on this information, which one of the following describes actions that the Unit Supervisor must direct, per the applicable procedure?

☐ **A** Verify the OPERABILITY of at least one Emergency Diesel Generators by performing the applicable Tech. Spec. surveillance.

☐ **B** Commence a plant downpower and secure all unnecessary equipment as the lower power permits.

☑ **C** Terminate any active Tech. Spec. surveillance test of any safety related pumps and motors and secure them, if possible.

☐ **D** Verify the OPERABILITY of the Steam Driven Auxiliary Feedwater Pump by performing the applicable Tech. Spec. surveillance.

Question Misc. Info: MP2*LOIT 2580, Degraded Voltage, Required Actions, MB-04720, NRC-2009 [U-SRO], NRC-2016

Justification

SRO ONLY QUESTION - Samples 55.43(1)(5) SRO needs to understand the Conditions and Limitation of the facility license and the assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

A - WRONG; This is required by the AC Power Source TSAS, 3.8.1.1, when an EDG is deemed not OPERABLE, but not when the inoperable AC Power Source is the off-site lines.
Plausible; Examinee may believe that pre-staging the EDGs would be a logical requirement as they will soon be needed to ensure safety functions are met upon loss of offsite power.

B - WRONG; AOP 2580 does not direct a plant down power be commenced as the loss of power to the grid is a far worse impact than any gains by securing equipment.
Plausible; Examinee may believe that because the applicable AOP directs that unnecessary loads be secured to help with the degraded voltage, and a trip from a lower power level is preferred, that lowering power to allow securing of components is logical.

C - CORRECT; To limit the risk of damage to safety related motor windings due to the higher current flows that would be expected, all unnecessary running of these components must be terminated.

D - WRONG; This is a requirement of TS 3.8.1.1 for loss of an EDG.
Plausible; Examinee may rationalize that on a probable trip from loss of the grid, one facility of vital AC power is always assumed to be lost due to some component or system failure. Therefore, in order to ensure two Aux. Feedwater pumps are available to meet the requirements of EOP 2525, Standard Post Trip Actions, the TDAFP must be verified to be OPERABLE.

SRO Justification: The question requires detailed knowledge of applicable TSAS and AOP required actions for degraded voltage.

References

AOP 2580

Comments and Question Modification History

Replaced all three distractors on original NRC approved question to improve SRO alignment. Question stem and correct answer were not changed.

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 2.2.17 **RO** 2.6 **SRO** 3.8 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.

Question #: **96** | **Question ID: 8000026** | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?
Rev. 1 | ☑ Selected for Exam | **Origin: Bank** | ☐ Past NRC Exam?

The plant was at 100% power when a Loss of Load caused the reactor to trip and the PORVs to briefly open. Fifteen minutes after the completion of EOP 2525, Standard Post Trip Actions, the following conditions are reported:

- RCS pressure is 2240 psia and stable.
- RCS temperature is 532°F and stable.
- 2-RC-402, "A" PORV, has dual indication
- Annunciator C-11 on C-02/3, PORV RC-402 OPEN is still lit.
- Quench Tank pressure is 15 psig and slowly rising.
- Quench Tank level is 56% and slowly rising at 1% every 5 minutes.
- Quench Tank temperature is 252°F and slowly rising.

Which of the following describes the minimum required response to these conditions?

☐ **A** Refer to the Diagnostic Flowchart, transition to EOP 2532, Loss of Coolant Accident and be in COLD SHUTDOWN within 36 hours.

☐ **B** Continue with EOP 2526, Reactor Trip Recovery, close the associated PORV Block Valve and be in COLD SHUTDOWN within 36 hours.

☑ **C** Continue with EOP 2526, Reactor Trip Recovery, close the associated PORV Block Valve or be in HOT SHUTDOWN within 6 hours.

☐ **D** Refer to AOP 2568, RCS Leakage, close and deenergize the associated PORV Block Valve, be in HOT SHUTDOWN within 6 hours.

Question Misc. Info: MP2*LOIT, QT, PZR, RCS, PORV, NRC-2008 [K/A; 008, 2.2.44], NRC-2016

Justification

SRO ONLY QUESTION - Samples 55.43(1)(5) SRO needs to understand the Conditions and Limitation of the facility license beyond 1 hour and the assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

A - WRONG; There is NO need to transition to EOP 2532; the conditions indicate approximately 3 gpm leakage past "A" PORV to the Quench Tank. This is only an RCS leak. Additionally, there is NO requirement to achieve COLD SHUTDOWN within 36 hours, although the crew may continue to MODE 5 to repair the leaking PORV.
Plausible; Examinee may focus on the **many** scenarios were a stuck open PORV was the cause of a LOCA and EOP 2532 is a viable succes path for the given conditions.

B - WRONG; The conditions presented are indicative of an RCS leak, NOT a LOCA; therefore, the crew should continue with EOP 2526; however, there is NO requirement to achieve COLD SHUTDOWN within 36 hours.
Plausible; Examinee may believe the PORVs have the similar Tech. Spec. Requirements as the PZR Safety Valves, in that they must be OPERABLE for the plant to go above < 275°F.

C - CORRECT; The conditions presented are indicative of an RCS leak, NOT a LOCA; therefore, the crew should continue with EOP 2526. Technical Specification LCO 3.4.3, ACTION a. requires the associated PORV Block Valve to be closed within 1 hour, with power maintained OR place the plant in HOT STANDBY within 6 hours and HOT SHUTDOWN within the next 6 hours. With the plant already in MODE 3, the crew has 6 hours to place the plant in MODE 4.

D - WRONG; Although NOT required, the crew may refer to AOP 2569. The associated PORV Block Valve must be closed, but there is NO need to deenergize it as long as the PORV can be manually cycled. Additionally, there is NO need to go to HOT SHUTDOWN as long as the deenergized associated PORV Block Valve is energized within 72 hours.
Plausible; Examinee may not take into account the need to possibly use the PORV to control RCS pressure at some later time, possibly with conditions below NOP/NOT, per the TS Bases.

References

TS 3.4.3, EOP 2541-APP01

Comments and Question Modification History

Modification from 2008 original: Added justification for SRO level and "Plausibility statements" to each distracter.

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

Generic K/A Selected

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

Question #: **96** **Question ID: 8000026** ☐ **RO** ☑ **SRO** ☐ **Student Handout?** ☐ **Lower Order?**

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

Number 2.2.44 **RO** 4.2 **SRO** 4.4 **CFR Link** (CFR: 41.5 / 43.5 / 45.12)

Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Question #: **97** | ***Question ID:*** ***1100052*** | ☐ ***RO*** | ☑ ***SRO*** | ☐ **Student Handout?** | ☐ **Lower Order?**

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

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

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

What action must be taken for this condition?

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

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

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

☐ **D** Per TSAS 3.0.3, within 1 hour, initiate ACTION to commence a plant shutdown and subsequent cooldown. Be in at least COLD SHUTDOWN with the subsequent 36 hours.

Question Misc. Info: MP2*LOIT Rad. Mon., SRO, NRC-2005 [K/A 061, ARM, AA2.01], NRC-2011, 55.43(b)(4), NRC-2016

Justification

SRO Justification; 10CFR55.43(1) SRO needs to understand the Conditions and Limitation of the facility license for in plant radiation monitors knowing the required Tech. Spec. action statement time limits beyond 1 hour

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

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

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

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

References

ARP 2590B-040, TS 3.3-6

NO Comments or Question Modification History at this time.

NRC K/A System/E/A **System** 2.3 Radiation Control

Generic K/A Selected

NRC K/A Generic **System** 2.3 Radiation Control

Number 2.3.5 **RO** 2.9 **SRO** 2.9 **CFR Link** (CFR: 41.11 / 41.12 / 43.4 / 45.9)

Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Question #: **98** **Question ID: 8000043** ☐ **RO** ☑ **SRO** ☐ **Student Handout?** ☑ **Lower Order?**
Rev. 0 ☑ **Selected for Exam** **Origin: Bank** ☐ **Past NRC Exam?**

A Fuel Handling Accident has occurred inside Containment. All personnel were immediately evacuated.

In order to limit personnel exposure, the Shift Manager must ensure which of the following are performed?

☐ **A** Containment must be isolated no more than 30 minutes after all personnel are evacuated from Containment.

☐ **B** The Containment Purge valves, AC-4, 5, 6, and 7 are required to be closed within 10 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 60 minutes of the event.

Question Misc. Info: MP2*LOIT, 2577, RM, FH, RF, CTMT, NRC-2008, NRC-2016

Justification

SRO Justification; 10CFR55.43(4)(5)(7) SRO assessment of radiation hazard that may arise during abnormal situations and understand the facility conditions and selection of appropriate procedures during normal, abnormal and emergency conditions and Fuel handling procedures to determine the correct actions for a fuel handling accident.

A - WRONG; Containment must be isolated with 30 minutes of the event, not after all personnel are out of Containment.
Plausible; Examinee may conclude it logical to wait for everyone to exit CTMT before attempting to isolate it.

B - WRONG; 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 been disabled, they must be manually closed immediately by a dedicated operator. The procedural allowed time delay for this action is 30 minutes or before the calculated core boiling time.
Plausible; Examinee may recall the valves are occasionally blocked open and would then need to be closed manually, with 10 minutes being the approximate minimum time to boil at the earliest possible start to fuel movement.

C - WRONG; 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. This valve would be closed for a loss of Refuel Pool level.
Plausible; Examinee may recall the valve is part of the CTMT closure requirements and mentioned in other AOPs dealing with fuel movement.

D - CORRECT; 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.

References

AOP 2577

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 2.3.13 **RO** 3.4 **SRO** 3.8 **CFR Link** (CFR: 41.12 / 43.4 / 45.9 / 45.10)

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:** **9079010** ☐ **RO** ☑ **SRO** ☐ Student Handout? ☐ Lower Order?

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

The plant was at 100% power when CONVEX ordered Main Generator output be lowered from 900 MWe to 600 MWe, 20 minutes ago.

AOP 2557, "Emergency Generation Reduction", was initiated and the following conditions now exist:

- Group 7 CEAs are at 170 steps withdrawn.
- Reactor power is 98% and slowly lowering.
- Main Generator output is 610 MWe and slowly lowering.
- "A" Steam Dump Bypass Valve is 75% open and stable.
- "BYPASS TO CND", PIC-4216 output is 80% and stable.
- "B", "C" and "D" Steam Dump Bypass Valves are open 75% and stable.
- "STEAM DUMP TAVG CNTL", TIC-4165 output is 85% and stable.
- "RC LOOP 1 COLD LEG TEMP HI" annunciator has just alarmed (C02/3; C-34).
- "RC LOOP 2 COLD LEG TEMP HI" annunciator has just alarmed (C02/3; D-34).
- RCS Tcold is 550 °F and slowly rising (RPS).

Which one of the following actions are appropriate per the applicable procedures?

☐ **A** Transfer control of the steam dumps to Foxboro IA control and restore Tcold to program.

☑ **B** Lower the setpoint on PIC-4216 or raise the output of TIC-4165 and log into the DNB TSAS.

☐ **C** Trip the plant, return controls to normal and go to EOP 2525, "Standard Post Trip Actions".

☐ **D** Stop generator load reduction and raise generator load until the C02/3 alarms have cleared.

Question Misc. Info: MP2*LOIT 2575, AOP, MB-05478, NRC-2009, NRC-2016

Justification

SRO Justification; 10CFR55.43(1) SRO needs to understand the Conditions and Limitation of the facility license. The SRO has to decide which is the most appropriate action to take when more than one mitigating action exists.

A - WRONG; Transferring the "A" steam dump to Foxboro IA control would immediately fail the valve closed, making things far worse. Plausible: This action is prudent if, upon initiation of the procedure, it was noted the controller on C05 was not responding properly. The indications given show the C05 controller may not be operating correctly, when it is quite possible the setpoint must be lowered to ensure the "A" steam dump stays ahead of the other 3 valves.

B - CORRECT; Turbine load has been lowered ahead of the "A" Steam Dump Valve controller setpoint, as indicated by TIC-4165 output being higher than PIC-4216. Therefore, raising the output of PIC-4216 to open the "A" steam dump or raising the output on TIC-4165 to open all four valves would both restore Temperature to program. This is an expected possible action if the setpoint on PIC-4216 is not lowered enough initially. However, Tcold is already above the DNB Tech. Spec. limit, so the LCO must be entered.

C - WRONG; AOP 2557 requires RCS temperature to be maintained within 10 °F of program or a plant trip is required. AOP 2557 maintains reactor power constant, therefore, Tcold should be ~545 °F per Attachment 1. RPS indication (and C02/3 alarms) indicate Tcold is >/=549 °F, which is < 10 °F above program value.
Plausible; If plant power level is extracted from the Main Generator output, then Tcold should be ~540 °F. This would mean that Tcold is > 10 °F above the program value and a trip is required.

D - WRONG; Stopping the load reduction and picking up load at this time would prevent the crew from meeting the time requirements of the procedure and CONVEX. In some instances, CONVEX could trip both Millstone units to meet line voltage limits.
Plausible: This is an acceptable action if temperature is out of band due to turbine load reduction being ahead of reactor power reduction.

References

AOP 2557

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

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

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

Question #: **100** | ***Question ID:*** ***89525*** | ☐ ***RO*** | ☑ ***SRO*** | ☐ **Student Handout?** | ☑ **Lower Order?**

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

2-MS-190A, "A" Atmospheric Dump Valve has just failed open.
S/G pressure is 850 PSIA.
The "A" ADV was taken to manual and will not close.

Which of the following describe the actions the US must direct to mitigate the ADV failure, and the order they would be directed?

☐ **A**
1. Lower turbine load to restore Tc to program with power <100%
2. Place "A" SGFP in manual
3. Place "A" FRV in manual
4. Restore SGFP speeds
5. Stabilize S/G levels

☐ **B**
1. Place both SGFPs in manual
2. Lower turbine load to restore Tc to program with power <100%
3. Restore SGFP speeds
4. Place "A" FRV in manual
5. Stabilize affected S/G level

☑ **C**
1. Place both SGFPs in manual
2. Place "A" FRV in manual
3. Lower turbine load to restore Tc to program with power < 100%
4. Restore SGFP speed
5. Stabilize affected S/G levels

☐ **D**
1. Place "A" and "B" FRVs in manual
2. Lower turbine load to restore Tc to program with power <100%
3. Place both SGFPs in manual
4. Restore SGFP speeds
5. Stabilize affected S/G level

Question Misc. Info: MP2*LOIT AOP, 2585, NRC-2016

Justification

SRO Justification; This scenario will cause an immediate impact to reactor power, SG water level and turbine load and therefore requires integrated plant knowledge and understanding of the "big picture" to properly mitigate. The actions taken must be directed within seconds and in a specific order, or the automatic response of plant systems will magnify the impact of the failure to the point where a plant trip is the outcome.

A - WRONG; Lower Turbine correct, place only “A” SGFP in manual is wrong
Plausible; The steam demand from the failed open ADV (equivalent to ~ 7%) will cause an immediate rise in reactor power above 100%. However, this extra demand is not seen by the #1 SG's WLC system, which will only respond to the dropping water level, this drop in S/G water level will be the most pressing parameter for the Operators due to the urgent need to restore to prevent a Reactor Trip. The other SG level will also be affected due to the loss of FRV DP. This requires both SGFPs be put in manual because both FRVs are moving.

B - WRONG; Place both SGFPs in manual BEFORE lowering turbine load is wrong.
Plausible; Extra steam demand is not seen by the #1 SG's WLC system, which will only respond to the dropping water level, this drop in S/G water level will be the most pressing parameter for the Operators due to the urgent need to restore to prevent a Reactor Trip. S/G feed pump react to the lowering S/G pressure causing SGFP speed to lower further causing a lowering of S/G water level the Operator’s first instinct would be to place SGFP speed to manual to prevent this.

C - CORRECT; The steam demand from the failed open ADV (equivalent to ~ 7%) will cause an immediate rise in reactor power above 100%. However, this extra demand is not seen by the #1 SG's WLC system, which will only respond to the dropping water level. Consequently, the SG feed pumps must be stopped from any further changes to input parameters before they make things worse. This must be done before changing FRV position to correct SG level, because both SGFPs are affected by the open ADV and this will impact both SGs, not just the one with the open ADV. Then a reduction in steam demand to reduce power to within license limits and before RPS trips the plant on high power. This will also help limit the magnitude of the impact on SGWL control, which is the next priority.

D - WRONG; Place both FRVs in manual is wrong, lower Turbine correct.
Plausible; S/G water level will be the most pressing parameter for the Operators due to the urgent need to restore to prevent a Reactor Trip the Examinee may believe that preventing a reactor trip by restoring S/G feed control would be the first priority. The other SG level will also be affected due to the loss of FRV DP. This requires both SGFPs be put in manual because both FRVs are moving. However, putting both valves in manual, although not "technically" wrong, would require pulling the RO into the mitigating strategy and leave no board operator monitoring the reactor (therefore "administratively" wrong).

References

AOP 2585

Question #: **100** **Question ID:** **89525** ☐ **RO** ☑ **SRO** ☐ **Student Handout?** ☑ **Lower Order?**

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

NO Comments or Question Modification History at this time.

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

Generic K/A Selected

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

Number 2.4.49 **RO** 4.6 **SRO** 4.4 **CFR Link** (CFR: 41.10 / 43.2 / 45.6)

Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.