1 | ☐ RO ☑ SRO | Question ID: 3000000 | Origin: New | ☐ Memory? (Check=Yes)

The plant is at 100% power steady-state with the following:
- Balanced charging & letdown with the backup selector switch in 1-2
- The Chg. Pp. Override switch is in Auto.
- Reactor Reg. Channel 'Y' is selected
- Pzr level controller L110Y is in service
- L110Y setpoint lever is selected to "R"
- Letdown flow control valve 2-CH-110P is in service
- Letdown backpressure control valve 2-CH-201P is in service

The feeder breaker for MCC B61 trips open while the PPO is at the back panels taking logs.
Nearly a minute has elapsed before he returns to the front panels.
There has been no operator actions taken on the primary side panels.

As the on-shift US, what would you direct the PPO to perform in response to this event?

A Shift Reactor Reg. to channel 'X' selected, caution operators regarding no Quick-Open signal ☐

B Restore charging and letdown using 'A' charging pump in lead ☑

C Take manual control of letdown using L110X, restore pzr level and return L110X to Auto control ☐

D Shift controller L110Y setpoint lever selected to "L" and adjust to regain proper level control ☐

Justification B: correct, with the stated initial conditions the 'C' chg pp was running, but trips on loss of B61, letdown isolates on high temperature, chg/LD must be restored using an alternate chg pp;
A: Rx Reg not lost, VR21 shifts to B62;
C: level controller L110Y retains power, i.e. no need to shift to L110X;
D: RRS stays powered so "Remote" setpoint is not effected and should not be selected to "Local" input.

Reference MP2*LOUT*CVCS, 2503F, MB-05663

NRC K/A System/E/A

System 022 Loss of Reactor Coolant Makeup

Number AA2.02

Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Pump Makeup: Charging pump problems

Importance RO/SRO 3.2 3.7

10CFR Link (CFR: 43.5 / 45.13)

NRC K/A Generic

# 2	☐ RO ☑ SRO	Question ID: 3074039	Origin: Bank	☐ Memory? (Check=Yes)

The plant is in MODE 5 with the following conditions:

* Shutdown Cooling System is in operation
* Shutdown cooling return temperature, T351X, is 140°F
* Both reactor coolant loops, with associated steam generator and one associated RCP, are OPERABLE

A loss of Shutdown Cooling occurs and RCS temperature reaches 235°F before SDC flow is restored.

Which of the following is a consideration regarding the cooldown to return the plant to the initial condition?

A Technical Specification cooldown limits may be waived when cooling down after an unplanned MODE change. ☐

B A one hour "soak" is required following a heatup of > 50 °F before initiating a cooldown. ☐

C The cooldown rate is limited to 80°F/hr until MODE 5 is reached, then 5°F/hr thereafter. ☐

D The cooldown rate is limited to 30°/hr regardless of the MODE change. ☑

Justification Per AOP 2572:
A) wrong; never allowed unless prescribed by EOPs
B) wrong; no such requirement exists anymore
C) wrong; see D.
D) correct; T.S. on cooldown limit when >230 is 80°F/hr; however, with the RCS at 235°F, the 30°F/hr limit is used to ensure the cooldown limit when <230°F is not violated.

Note: The C/D Rate Tech. Spec. 3.4.9.1, Table 3.4-2 MUST be removed from referenced material given to Initial License Examinees.

Reference MP2*LORT SDC, 2572, MB-05840
Note: The C/D Rate Tech. Spec. 3.4.9.1, Table 3.4-2 MUST be removed from referenced material given to Initial License Examinees.

NRC K/A System/E/A		NRC K/A Generic
System	025 Loss of Residual Heat Removal System (RHRS)	Conduct of Operations
Number	GA	2.1.33
	SEE GENERIC K/A	Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.
Importance RO/SRO		3.4 4.0
10CFR Link		(CFR: 43.2 / 43.3 / 45.3)

# 3	☐ RO ☑ SRO	Question ID: 3000001	Origin: New	☐ Memory? (Check=Yes)

The following plant conditions exist:

* 100% power
* All systems are in normal operation
* Forcing PZR Sprays for boron equalization

Then, VR-21 is lost causing all Backup Heaters to trip. RCS pressure drops to 1995 psia before turning due to manipulations of the remaining available pressure control components. VR-21 remains deenergized.

Which of the following describes an applicable action for the above plant conditions?

A Restore RCS pressure to within its limits within two (2) hours. ☑

B Restore at least two (2) groups of Backup heaters within 72 hours. ☐

C Be in HOT STANDBY within one (1) hour. ☐

D Restore the inoperable bus to operable status within eight (8) hours. ☐

Justification A; CORRECT - RCS pressure has dropped below the DNB minimum requirements and must be restored within 2 hours.
B; WRONG - 3.4.4 requires at least two groups of heaters, but they must be vitally powered. Therefore, only the Proportional Heaters are applicable to this Tech. Spec.
C; WRONG - Although the pressure is below one of the minimum pressure lines on the Safety Tech. Spec., the corresponding power does not align properly with this minimum pressure on the "unacceptable" side of the line.
D; WRONG - Although the "backup" power supply to a Tech. Spec. required vital power supply has been lost, this does not, in and of itself, make the applicable vital power supply inoperable.
Requires the use of Technical Specifications

Reference MP2*LOUT, TS, PPLCS, VR-21, MB-04978

NRC K/A System/E/A		NRC K/A Generic
System	027 Pressurizer Pressure Control System (PZR PCS) Malfunction	Equipment Control
Number	GA	2.2.22
	SEE GENERIC K/A	Knowledge of limiting conditions for operations and safety limits
Importance RO/SRO		3.4 4.1
10CFR Link		(CFR: 43.2 / 45.2)

# 4	☐ RO	☑ SRO	Question ID: 3000002	Origin: New	☐ Memory? (Check=Yes)

The unit is at 100% power, steady state, when a number of alarms come in on C08, including: "Inverter INV-1 Trouble" and "Inverter INV-5 Trouble".
Channel 'A' of safety systems' instrumentation and channel 'A' of the RPS are de-energized.
Four TCBs have tripped.

Based on the above conditions, which of the following describes one of the actions the US should take?

A Direct both Main Feedwater Pumps be placed in MANUAL speed control. ☐

B Direct Charging and Letdown be immediately secured. ☑

C Direct Pressurizer Heater Control be selected to Channel "X". ☐

D Direct the performance of EOP-2525, Standard Post Trip Actions. ☐

Justification A; WRONG - Loss of VA-10 locks up the main feed reg. Valve, pump control is VA-30 and VR-11.
B; CORRECT - System is now Emergency Borating with a locked up feed reg. Valve. Will quickly become a very unstable situation.
C; WRONG - PLPCS should be selected to Channel "X" to regain control, heater control swith will NOT recover lost PZR heaters.
D; WRONG - The plant can trip with the loss of four breakers (cause of March '03 trip), but the ones lost due to VA-10 deenergizing would NOT cause a plant trip due to the TCB ring bus configuration.

Reference MP2*LOUT, VIAC, 2504C (CFR 55.43.b.2), MB-5750

NRC K/A System/E/A

System 057 Loss of Vital AC Electrical Instrument Bus

Number AA2.06

Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: AC instrument bus alarms for the inverter and alternate power source

Importance RO/SRO 3.2 3.7

10CFR Link (CFR: 43.5 / 45.13)

NRC K/A Generic

# 5	☐ RO ☑ SRO	Question ID: 3100006	Origin: New	☐ Memory? (Check=Yes)

With the plant operating at full power, the following plant conditions exist:
- Letdown flow is 29 gpm.
- Charging flow is 88 gpm.
- RCS Tavg is 573 °F, steady.
- Pressurizer level is 63%, lowering.
- Steam Generator levels are 70%, steady.
- Steam Generator pressures are 870 psia, steady.
- Containment pressure is 0.0 psig, steady.
- Containment temperature is 112 °F, steady.
- Containment sump level is 35%, steady.
- The Steam Jet Air Ejector Radiation Monitor alarm has just been received.
- The N-16 Radiation Monitor alarm has just been received
- "B" Steam Line Radiation Monitor alarm has just been received.
- All other rad monitor readings are normal.
- The wind is blowing from 12°, to 192°, at 12 mph.

Due to the above conditions, the US orders the plant tripped and EOP 2525 carried out. During the performance of EOP 2525, the SPO reports that a Main Steam safety valve on the #2 S/G has stuck partially open. The crew completes EOP 2525 and enters EOP 2540.

What direction will the SM provide to all on-site personnel?

A On-shift operators will report to the Unit 2 Control Room. All SERO members will stand by for further instructions. All other personnel will continue with their present duties. ☐

B All SERO members will report to the alternate Emergency Response Facility on site. All other personnel will proceed to the appropriate access point for 'early dismissal'. ☐

C All SERO members will report to their designated Emergency Response Facility. All other personnel will 'evacuate' the site while Security performs an accountability. ☑

D Station personnel NOT on shift will evacuate the site. All SERO members will assemble in the EOF. All other personnel will evacuate the site to a distance of at least 5 miles. ☐

Justification C is correct. The stated conditions require a classification of a Site Area Emergency. As a result, SERO must be activated and an evacuation and accountability must be initiated per MP 26-EPI-FAP08, Evacuation and Accountability.
A is incorrect. This would be appropriate for an Unusual Event.
B is incorrect. This would be appropriate if the radiation release was going to challenged the dose limits within the EOF. This is NOT the case due to plum source or direction of travel.
D is incorrect. This MAY be appropriate for an event which would endanger personnel on site who are NOT protected in the Control Room.

Reference MP2*LOUT, MP-26, 2525, 2540, SRO, SGTR, EAL, ADMIN, MB-01471

Note: EALs and a Site Map (with compus reference) must be provided.

NRC K/A System/E/A		NRC K/A Generic
System	E09 Functional Recovery	Conduct of Operations
Number	GA	2.1.14
	SEE GENERIC K/A	Knowledge of system status criteria which require the notification of plant personnel.
Importance RO/SRO		2.5 3.3
10CFR Link		(CFR: 43.5 / 45.12)

# 6	☐ RO ☑ SRO	Question ID: **3056699**	Origin: **Modified**	☐ Memory? (Check=Yes)

The plant is operating at 100% power, steady state, with Channel "Y" selected as the CONTROLLING channel for pressurizer level control.
Then, the PPO notices the following RCS and Pressurizer (PZR) indications:

- o All Backup heaters are on.
- o The Proportional heaters are at maximum.
- o Channel "Y" indicated PZR level is approximately 69% and slowly rising.
- o Channel "X" indicated PZR level is at 58% and going down.
- o Cold Calibrated PZR level is at 47% and slowly rising.
- o PZR pressure is slowly going down.
- o RCS temperature is constant.

Which of the following describes a possible cause for these indications, and the action that should be directed by the Unit Supervisor (US)?

A A small leak has developed in the REFERENCE leg of Channel "Y" of Pressurizer Level Control. Direct the PZR level controls and PZR heater selector switch be shifted to Channel "X" control. ☑

B A small leak has developed in the VARIABLE leg of Channel "X" of Pressurizer Level Control. Direct the PZR heater selector switch be shifted to Channel "Y" only. ☐

C Tavg from the controlling Reactor Regulating system channel (Y) has failed LOW. Direct the Reactor Regulating system be shifted to the unaffected channel (X). ☐

D Letdown Back Pressure Controller is failing, causing a loss to the system through the letdown relief valves.
Direct taking manual control of the Back Pressure controller and restoring normal letdown flow. ☐

Justification A; CORRECT - a leak in the reference leg to channel "Y" would cause both Channel "Y" and the Cold Calibrated level indications to rise due to a shared reference leg. Shifting BOTH switches is required by procedure due to the failing channel controls.
B; WRONG - A leak in the variable leg of Channel "X" would cause "X" indication to drop, but would not cause both of the other level indications to rise.
C; WRONG - RRS Tavg failing low would cause actual PZR level to lower, due to a lowering setpoint, but would not cause differing level indications between channels.
D; WRONG - When the back pressure controller failed in early 2003, it caused INDICATED letdown flow to go to zero, but letdown never really changed because the letdown relief valve had opened, diverting letdown to Rad. Waste. Although there is a mass loss to the system, PZR level does not initially see it because charging and letdown flow have not actually changed.

Reference MP2*LOUT, 2304A, PLPCS, MB-2314

NRC K/A System/E/A

System 028 Pressurizer (PZR) Level Control Malfunction

Number AA2.10

Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: Whether the automatic mode for PZR level control is functioning improperly, necessity of shift to manual modes

Importance RO/SRO 3.3 3.4

10CFR Link (CFR: 43.5 / 45.13)

NRC K/A Generic

7 | ☐ RO ☑ SRO | Question ID: 3100125 | Origin: Bank | ☐ Memory? (Check=Yes)

The plant is operating at 90% power with the "C" circulating pump out of service for maintenance.

Then, the following conditions are observed and reported:

- TRAVELING SCREEN DELT-P HI annunciator D-10 on C-06/7
- "D" traveling screen delta-P suddenly rose to 40 inches and subsequently dropped to 2 inches.
- Condenser vacuum is 2.9 inches Hg abs. and slowly getting worse.
- CIRC WATER PUMP D OVERLOAD/TRIP, annunciator D-9 on C-06/7

Which of the following actions will the US direct?

A Cross tie the "D" water box with the "B" water box. ☐

B Trip the reactor and Go To EOP 2525. ☑

C Place the mechanical vacuum pumps in service. ☐

D Reduce power at the maximum attainable rate until vacuum stabilizes. ☐

Justification "B" is correct. The given conditions indicate a loss of the "D" circulating water pump. With the "C" circ water pump out for maintenance, this constitutes a loss of two waterboxes in one condenser. According to AOP 2517, a reactor trip is required. (This question relates to a previous event on Unit 2.)
"A" is incorrect because the annunciator response procedure does NOT require cross tying water boxes. Additionally, the "B" and "D" water boxes CANNOT be crosstied.
"C" is incorrect because AOP 2574 does NOT provide operational guidance for the loss of two water boxes in one condenser; however, it does reference AOP 2517 for the loss of more than one circulating water pump in the AOP entry conditions.
"D" is incorrect because although AOP 2575 provides guidance to reduce power for a deteriorating condenser vacuum, it does NOT specifically include the loss of two water boxes in one condenser as an entry condition.

Note: K/A Catalog links question to 55.43.2, however, question also meets 55.43.5 criteria. (NRC input)

Reference MP2*LORT, SRO, CWS-010C, CW, AOP 2517, CFR555.43(b)(5), MB-06119

NRC K/A System/E/A		NRC K/A Generic
System	051 Loss of Condenser Vacuum	Emergency Procedures /Plan
Number	GA	2.4.49
	SEE GENERIC K/A	Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.
Importance RO/SRO		4.0 4.0
10CFR Link		(CFR: 41.10 / 43.2 / 45.6)

8 | ☐ RO ☑ SRO | Question ID: 3071905 | Origin: Modified | ☐ Memory? (Check=Yes)

The plant has experienced a Loss of Coolant Accident and the following conditions exist:

- Sump Recirculation has occurred.
- The Safety Injection Recirculation Header Isolation valves, 2-SI-659 and 660, are the ONLY SRAS actuated components that have NOT automatically positioned. All other SRAS actuations have occurred as designed.

Which one of the following statements describes WHEN the Unit Supervisor would direct these valves to be closed and WHY?

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

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

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

D Only after overriding and securing both LPSI pumps to ensure the loss of minimum flow does not damage the HPSI and CTMT Spray pumps if secured with all three running. ☐

Justification A; WRONG - These are not the valves that would "short-cycle" the CS through HPSI and closing them is listed as the *SIXTH step in the "Supplemental Actions" following a SRAS.*
B; WRONG - HPSI flow must be verified to be 30 gpm, and actions to secure a HPSI pump must be taken if it is less than that, NOT leaving the minimum flow recirc path open.
C; CORRECT - This is in the initial actions following a SRAS and must be verified or manually accomplished to ensure a direct release to the environment does not exist.
D; WRONG - The LPSI pumps should have automatically been secured, as stated in the stem's amplifing information, they do not have to be overridden. Running these pumps during a SRAS for any length of time would rob the HPSI and CS pumps of vital NPSH due to the hotter suction source.

Reference MP2 LOUT, EOP 2532, LOCA, MB-04749

NRC K/A System/E/A

System 013 Engineered Safety Features Actuation System (ESFAS)

Number A2.01

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

Importance RO/SRO 4.6 4.8

10CFR Link (CFR: 41.5 / 43.5 / 45.3 / 45.13)

NRC K/A Generic

# 9	☐ RO ☑ SRO	Question ID: 3010068	Origin: Bank	☐ Memory? (Check=Yes)

The main turbine has just been rolled to 1800 rpm during a plant startup.
The PPC has just completed its normally scheduled run of the INPAX program.
The "Fr-T - Tq" alarm on panel C04 annunciates and locks in.
The STA reports that the INPAX value for Fr-T is 1.83 and Tq is 0.04.

Based on the above, the US will direct which of the following actions?

A Halt the plant startup until RE provides a Reactivity Management Plan for CEA movement to correct power distribution. ☐

B Reduce thermal power to bring thermal power and Fr-T to within the limit specified in the COLR. ☐

C Correct the power tilt within 2 hours and determine within the next 2 hours that Fr-T is within the required limit. ☐

D Continue the plant startup, manually run INPAX when power exceeds 20% and verify acceptable value for Fr-T. ☑

Justification A; WRONG - chosen if examinee assumes power distribution data is valid and must be corrected by 'smoothing' core flux distribution.
B; WRONG - chosen if examinee neglects power level determination or TS applicability for Fr-T, but assumes <50% for Tq.
C; WRONG - chosen if examinee neglects power level determination or applicability for both parameters.
D; CORRECT - examinee must first determine power level (turbine roll ~13% power), Fr-T value not considered valid until core power >20% and Tq not spec'd until >50% power.
Requires the use of Technical Specifications and Figure 2.6-1 of the COLR

Reference MP2*LOIT/LOUT, 2203, TS, (CFR-55.43.b.2, 5), MB-5405

NRC K/A System/E/A		NRC K/A Generic
System	015 Nuclear Instrumentation System	Equipment Control
Number	GS	2.2.22
	SEE GENERIC K/A	Knowledge of limiting conditions for operations and safety limits
Importance RO/SRO		3.4 4.1
10CFR Link		(CFR: 43.2 / 45.2)

# 10	☐ RO ☑ SRO	Question ID: 3073874	Origin: Bank	☐ Memory? (Check=Yes)

A reactor shutdown for refueling is in progress with CEA Groups 6 and 7 being inserted in Manual Sequential.

Which of the following describes when the US is allowed to log into MODE 3?

A When Regulating CEAs have been inserted enough to declare the reactor subcritical. ☐

B Only after RCS boron concentration has been verified to be greater than or equal to the shutdown requirement. ☐

C Only after all Regulating CEAs and Shutdown CEAs are FULLY inserted into the core. ☐

D When all Regulating CEAs in Group 4 are at or below the minimum Transient Insertion Limit. ☑

Justification A; WRONG - This is implied in the Tech. Spec. definition for Mode 3, but is NOT as restrictive as those defined in the Reactor Shutdown procedure.
B; WRONG - This is required for "refueling" operations, but is NOT defined as the point of Mode change.
C; WRONG - The Reactor Shutdown procedure alternate method of reactor shutdown has the operators "simultaneously insert" all CEAs (manual trip), then log into Mode 3. But this is NOT the "defined" transition point for entry into Mode 3 when manually inserting CEAs.
D; CORRECT - per OP-2206, Reactor Shutdown procedure, Rev. 10, Step 4.2.2c, states to log into Mode 3 "when Group 4 is at 72 steps" - which, per the COLR, is the minimum Transient Insertion Limit.

Reference MP2*LORT 2206, MODE, MB-05385

NRC K/A System/E/A		NRC K/A Generic
System	2.1 Conduct of Operations	Conduct of Operations
Number	G	2.1.2
	SEE GENERIC K/A	Knowledge of operator responsibilities during all modes of plant operation.
Importance RO/SRO		3.0 4.0
10CFR Link		(CFR: 41.10 / 45.13)

11 | ☐ RO ☑ SRO | Question ID: 0053345 | Origin: Bank | ☐ Memory? (Check=Yes)

The plant is operating at 100% power. Based on the following information, what is the minimum level necessary for each BAST to, BY ITSELF, conform with the Technical Specification requirement?

- o RWST concentration = 1.09% (1900 ppm)
- o RWST level = 95% (451,250 gal)
- o "A" BAST concentration = 3.25% (5683 ppm)
- o "B" BAST concentration = 3.00% (5245 ppm)

A	'A' BAST min level 3750 Gals	'B' BAST min level 4878 Gals	☐
B	'A' BAST min level 4050 Gals	'B' BAST min level 5245 Gals	☐
C	'A' BAST min level 4600 Gals	'B' BAST min level 5450 Gals	☑
D	'A' BAST min level 4350 Gals	'B' BAST min level 5200 Gals	☐

Justification ATTACH Pg. 3/4 1-17 of TS to EXAM.
By referencing Tech Spec Figure 3.1-1 the volumes necessary to be above the curve for 1900 ppm RWST concentration, given the concentration of the A and B BASTs at 3.25% and 3.0% respectively requires at least 4600 and 5245 gal respectively.

REF: T.S. 3.1.2.8 T.S. FIG 3.1-1

Reference MP2*LORT*2318 [004 CVC-01-C 3112] (9/30/97) CVCS, TS 3.1.2.8, ADMIN, RWST, BAST

NRC K/A System/E/A		NRC K/A Generic
System	2.1 Conduct of Operations	Conduct of Operations
Number	G	2.1.25
	SEE GENERIC K/A	"Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data."
Importance RO/SRO		2.8 3.1
10CFR Link		(CFR: 41.10 / 43.5 / 45.12)

12 | ☐ RO ☑ SRO | Question ID: 3000005 | Origin: New | ☐ Memory? (Check=Yes)

The plant is at 100% power, steady state. I&C technicians have a work order to allow testing and calibration of the Pressurizer Proportional Heater's control SCRs. This test would require the BOTH facilities of Proportional Heaters be temporarily turned off while the control circuit for the SCRs is inspected.

Which one of the following describes a possible impact of allowing this test to be performed at this time, based on any Technical Specification requirements of the components involved in the test?

A On a sudden pressurizer insurge due to loss-of-load, a sufficient pressurizer steam bubble may not be maintained to prevent the RCS from going solid. ☐

B On a plant trip coincident with a Loss-Of-Offsite-Power (LNP), RCS pressure may not be sufficient to maintain natural circulation in Hot Standby. ☑

C On the expected pressurizer outsurge during the design basis Excess Steam Demand event, RCS pressure may be insufficient to ensure voiding does not prevent RCS flow and core cooling. ☐

D On a Steam Generator Tube Rupture event, pressurizer heater input may not be sufficient to prevent the isolated steam generator from inhibiting RCS depressurization. ☐

Justification A; WRONG - This is the basis for pressurizer maximum level, which although true, is not impacted sufficiently by proportional heater loss.
B; CORRECT - This is the basis for the Proportional Heaters and what will be directly impacted if they are rendered inoperable.
C; WRONG - Although the Design Basis ESD may substantially drop PZR level, the proportional heaters do not have the capacity to overcome the loss. Also, the loss of PZR level is due to shrinkage from overcooling, which eliminates the concern for insufficient core cooling.
D; WRONG - This is a concern in a SGTR event, but it is addressed by the cooling of the effected SG directly. The return of RCS pressure control to the PZR is desirable, but not the inherent solution.

Reference MP2*LOUT, TS, Bases, PZR, MB-04806

NRC K/A System/E/A		NRC K/A Generic
System	2.2 Equipment Control	Equipment Control
Number	G	2.2.25
	SEE GENERIC K/A	Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.
Importance RO/SRO		2.5 3.7
10CFR Link		(CFR: 43.2)

# 13	☐ RO ☑ SRO	Question ID: 3073950	Origin: Bank	☐ Memory? (Check=Yes)

Which of the following is NOT the responsibility of the Shift Manager/Unit Supervisor during the process of reviewing or installing a Temporary Modification?

A Determine whether a 50.59 Screen or a 50.59 Evaluation is required. ☑

B Determine whether the verification method will be visual inspection or functional check. ☐

C Ensure that necessary technical specification requirements are implemented. ☐

D Ensure that the Temporary Modification is compatible for the existing plant and system conditions. ☐

Justification A, B, and C are responsibilities of SM/US when reviewing Temp. Mods. for installation per WC-10, Temporary Modifications.
D is the responcibility of the System Engineer.

Reference MP2 LORT, 119 ADM WC 10, MB-4802

NRC K/A System/E/A		NRC K/A Generic
System	2.2 Equipment Control	Equipment Control
Number	G	2.2.14
	SEE GENERIC K/A	Knowledge of the process for making configuration changes.
***Importance* RO/SRO**		2.1 3.0
10CFR Link		(CFR: 43.3 / 45.13)

# 14	☐ RO	☑ SRO	Question ID: 3053375	Origin: Bank	☐ Memory? (Check=Yes)

While preparing for an Aerated Waste Discharge, the PEO reported to the Shift Manager that during the initial setup of the Aerated Waste System radiation monitor (RM-9116) the radiation monitor appears to be failed such that the discharge isolation valves will NOT close on a high radiation alarm. I&C troubleshooting reveals a three (3) day repair/replace time for the control circuit.

Which one of the following actions should the SM direct in order to COMPLETE this discharge?

A A SECOND sample must be drawn and analysis results verified by a SECOND Chemist before the tank may be discharged. ☑

B The Chemist must resample the tank every half hour during the discharge to ensure compliance of limits based on the maximum allowable by 10CFR20. ☐

C The Aerated Waste Monitor Tank must be discharged at a flow rate based on the quantity of the radioactive isotope with the GREATEST concentration. ☐

D I&C must repair or replace the radmonitor BEFORE the Aerated Waste Monitor Tank may be discharged. ☐

Justification A; CORRECT - SP 2617A; If the radmonitor is incapable of closing the discharge valves on a Hi Rad., it must be considered inoperable and the applicable actions taken. Tech. Specs. require a second sample be drawn and calculated by a second Chemist for an inop. rad. monitor.
B; WRONG - There is no difference in the way the "limits" are calculated when the rad. monitor is inop.
C; WRONG - ALL isotopes must be considered when calculating a discharge permit, regardless of the rad. monitor status.
D; WRONG - It is not necessary to repair/replace the rad. monitor in order to perform the discharge.

Reference MP2*LORT*2513 [069 PIO-04-C 4799] (11/21/97) 2617A, 2617, ARW, RM, MB-04814

NRC K/A System/E/A		NRC K/A Generic
System	2.3 Radiation Control	Radiation Control
Number	G	2.3.11
	SEE GENERIC K/A	Ability to control radiation releases.
Importance RO/SRO		2.7 3.2
10CFR Link		(CFR: 45.9 / 45.10)

# 15	☐ RO ☑ SRO	Question ID: 3100054	Origin: Bank	☐ Memory? (Check=Yes)

A plant cooldown is in progress using OP 2207.
The protected facility is Facility 1.
RCS temperature is 270° F and pressure is 375 psia with the 'A' & 'B' RCPs running.
The Shutdown Cooling System is in recirc for the warmup/pressurization leak check using the 'B' LPSI pump.
The 'A' HPSI pump is OPERABLE per applicable procedures and the applicable Tech Spec.

The Seismic Monitor alarm comes in accompanied by a strong ground movement.
RCS pressure drops rapidly with pressurizer level going off-scale low and containment pressure rapidly rising to 6 psig.

Based on the initial conditions and this event, what actions must the US direct?

A Ensure the 'A' HPSI and 'A' LPSI Pumps have automatically started and their associated injection valves automatically have opened. Ensure all CAR fans are running in slow speed. ☐

B Observe automatic operation of the 'A' HPSI Pump and verify all CAR fans have automatically started in or shifted to slow speed. Secure SDC and realign LPSI for injection. ☐

C Monitor automatic operation of all CAR fans in slow speed. Remove 'A' HPSI Pump handswitch from Pull-To-Lock and monitor its start. Secure SDC and realign LPSI for injection. ☑

D SIAS was procedurally blocked after reaching 1750 psia in the RCS. Manually actuate SIAS via C01 push buttons and verify HPSI Pumps, LPSI Pumps and CAR Fans start and injection valves align. ☐

Justification C: correct, SIAS on CTMT pressure can't be blocked so CAR fans will respond accordingly, the 'A' HPSI is required to be in PTL due to lowered RCS temperature and LPSI alignment prevents injection until realigned, these actions are directed by OP 2207 in response to a LOCA.
A: chosen by examinees if they key on Facility 1 protected and believe HPSI and LPSI are maintained fully operable.
B: chosen by examinees if they forget that HPSI is procedurally placed in Pull-To-Lock for PTS.
D: chosen by examinees if they forget that SIAS will actuate on CTMT pressure and LPSI needs to be manually aligned.

Reference MP2*LOIT/LOUT, LBLOCA, 2207, manual ESAS, (CFR 55.43.b.5), MB-05326

NRC K/A System/E/A		NRC K/A Generic
System	2.4 Emergency Procedure /Plan	Emergency Procedures /Plan
Number	G	2.4.9
	SEE GENERIC K/A	Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies.
Importance RO/SRO		3.3 3.9
10CFR Link		(CFR: 41.10 / 43.5 / 45.13)

# 16	☐ RO ☑ SRO	Question ID: 3100002	Origin: New	☐ Memory? (Check=Yes)

The plant is operating at 100% power when the SPO reports that Instrument Air header pressure is at 95 psig and lowering. The Turbine Building PEO reports a large unisolable leak just upstream of 2-IA-25, Turbine Building Instrument Air Isolation.

Assuming Instrument Air header pressure continues to lower, at what pressure in the Instrument Air System must (as directed by procedure) the US direct a manual reactor trip and why?

A Prior to reaching 90 psig; At 90 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 high moisture content of Station Air. ☐

B When pressure lowers to below 85 psig; At 85 psig the crew is procedurally directed to crosstie air with Unit 3. Continued operation in this alignment would result in all components supplied by Instrument Air to be inoperable, which is an unanalyzed condition. ☐

C When pressure lowers to below 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 is questionable. ☑

D Prior to reaching 60 psig; The Auxiliary Feed Regulating Valves will lock up with less than 60 psig supply pressure. The reactor must be tripped to allow the initial automatic opening of these valves and begin feeding Steam Generators. ☐

Justification C is correct. 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.
A is incorrect. Although the Instrument Air/Station Air Crosstie valve opens, continued operation with Station Air cross tied to Instrument Air is acceptable.
B is incorrect. Although is less desirable to operate with Unit 2 cross tied with Unit 3, there are NO restrictions; therefore No requirements to trip.
D is incorrect. The Auxiliary Feed Regulating valves has back up air that will ensure their operation for a limited duration even during a complete loss of Instrument Air.

Note: K/A Catalog refers to 55.43.2, however, the question also applies to 55.43.5 (NRC input)

Reference MP2*LOUT, IA, OP 2332B, AOP 2563, MB-04688

NRC K/A System/E/A		NRC K/A Generic
System	065 Loss of Instrument Air	Conduct of Operations
Number	GA	2.1.32
	SEE GENERIC K/A	*Ability to explain and apply all system limits and precautions*
Importance RO/SRO		3.4 3.8
10CFR Link		(CFR: 41.10 / 43.2 / 45.12)

# 17	☐ RO ☑ SRO	Question ID: 3100003	Origin: New	☐ Memory? (Check=Yes)

You are the Unit Supervisor on the night shift. The plant is operating normally at 100% power. A student in training for an RO license is standing a watch under the supervision of the Secondary Plant Operator (SPO).

The Primary Plant Operator (PPO) suddenly becomes ill and has to leave. A third (work control) SRO is NOT available. A replacement is called and will report to work in about 30 minutes. Five minutes after the PPO leaves, the plant experiences a loss of DV-20.

Which of the following statements, as allowed by procedure, describes your responsibilities for this event?

A Request the Shift Manager to perform the duties of the PPO while you maintain a command and control function. ☐

B Direct the SPO trainee to perform the required actions on the primary plant under your direct supervision. ☐

C Direct the Station Duty Officer to implement the appropriate procedure(s) while you perform the required actions of the PPO. ☐

D Perform the required actions of the PPO until properly relieved while the Shift Manager maintains a command and control function. ☑

Justification D is correct. OP 2260, step 1.21.5, states, "In the event it becomes necessary to provide assistance to the board operators, this assistance should be provided by the US in all cases. The Shift Manager should maintain his overall perspective of plant conditions." *In this case, a loss of DV-20 will result in a plant trip due to closure of the #2 MSIV.* The US should perform the actions of the PPO until relieved.
A is incorrect because the SM should always 'maintain his overall perspective of plant conditions.' (OP 2260, step 1.21.5)
B is incorrect because the trainee can ONLY 'operate controls with permission of the SM/US and under the direct supervision of a qualified Operator.' The US cannot maintain direct supervision of the trainee during the performance of EOP 2525. Additionally, trainee operation of equipment must be immediately suspended during unanticipated or abnormal events... (MP-14-OPS-GDL300)
C is incorrect because the Station Duty Officer, by definition, is NOT licensed on Unit 2 and CANNOT perform licensed duties.

Reference MP2*LOUT, AOP 2506B, DC, OP 2260, MB-06199

NRC K/A System/E/A		NRC K/A Generic	
System	058 Loss of DC Power	Conduct of Operations	
Number	GA	2.1.2	
SEE GENERIC K/A		Knowledge of operator responsibilities during all modes of plant operation.	
Importance RO/SRO		3.0	4.0
10CFR Link		(CFR: 41.10 / 45.13)	

# 18	☐ RO ☑ SRO	Question ID: 3100004	Origin: New	☐ Memory? (Check=Yes)

The plant is shut down for a refueling outage with fuel movement in progress. Channel "C" Wide Range instrument is inoperable to allow for the scheduled replacement of the detector. Additionally, several incore detectors are being replaced at this time. Channels "A", "B", and "D" are in operation with Channel "B" providing audible indication.

Then, Channel "B" Wide Range indication is suddenly lost due to a blown fuse.

Which of the following statements describe the appropriate response?

A Fuel movement may be directed to continue. The required number of Source Range Neutron Flux Monitors are still OPERABLE. ☐

B Immediately suspend fuel movement, select either channel "A" or "D" for audible indication, and recommence fuel movement. ☑

C Fuel movement may be directed to continue provided the SHUTDOWN MARGIN requirements of Specification 3.1.1.2 are immediately verified. ☐

D Immediately demonstrate the OPERABILITY of a redundant remote shutdown channel or restore the inoperable channel to OPERABLE within 7 days. ☐

Justification B is correct. Although two of the required Source Range Neutron Flux Monitors are still OPERABLE, the audible count rate is inoperable. This will require the immediate suspension of CORE ALTERATIONS until the audible count rate is restored.
A is incorrect because the audible count rate in the control room and in containment has been lost; therefore CORE ALTERATIONS must be immediately suspended. CORE ALTERATIONS may be restarted when the audible count rate is restored.
C is incorrect. The requirements of Spec 3.1.1.2 are only applicable in MODE 5. (The plant must be in MODE 6 to allow incore detector replacement.)
D is incorrect because the requirements for Remote Shutdown Instrumentation are NOT applicable in MODE 6.
** Requires the use of Technical Specifications **

Reference MP2*LOUT, TS, NIS, 2380, MB-01441

NRC K/A System/E/A		NRC K/A Generic	
System	032 Loss of Source Range Nuclear Instrumentation	Equipment Control	
Number	GA	2.2.22	
SEE GENERIC K/A		Knowledge of limiting conditions for operations and safety limits	
Importance RO/SRO		3.4	4.1
10CFR Link		(CFR: 43.2 / 45.2)	

# 19	☐ RO	☑ SRO	Question ID: 3100014	Origin: New	☐ Memory? (Check=Yes)

A large electrical fire has been reported in the Unit 2 Auxiliary Building 25' Cable Vault. Heavy smoke is quickly filling the control room and the SM has determined that this is an Appendix "R" fire. He then orders the control room evacuated and the applicable Appendix "R" actions followed.

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

A ☐
1. Evacuate the Control Room
2. Trip the reactor from the 480 VAC switchgear rooms
3. Place all switches on the Bottle-Up Panels in the "ISOL" position
4. Establish feed to #2 Steam Generator from C21.

B ☑
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. Place all switches on the C10 Panel to the "LOCAL" position.
4. Operate the #2 ADV to control RCS temperature.
5. Establish feed to #2 Steam Generator with the Terry Turbine

C ☐
1. Trip the reactor
2. Place #2 ADV controller in Manual/Close
3. Establish feed to #2 Steam Generator with the Terry Turbine
4. Evacuate the Control Room
5. Place all switches on the Bottle-Up Panels in the "ISOL" position

D ☐
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 feed to either Steam Generator from C21 using Aux. Feed.
4. Place either ADV controller in Manual and control RCS temperature from C10

Justification B is correct. AOP 2579A lists the required actions in the following order: 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.
A, WRONG; C21 Panel is NOT used during an Appendix "R" fire even though it allows for better plant control.
C. WRONG; The TDAFP is NOT operated from the control room as its control wiring is assumed to be shorted out in this type of Appendix "R" fire.
D. WRONG; C21 can feed either SG, but can NOT be trusted to do so during this type of Appendix "R" fire. Also, C10 panel can only control the #2 ADV.

Reference MP2*LOUT, Fire, 2579A, MB-04722

NRC K/A System/E/A		NRC K/A Generic	
System	2.4 Emergency Procedure /Plan	Emergency Procedures /Plan	
Number	G	2.4.25	
SEE GENERIC K/A		Knowledge of fire protection procedures.	
Importance RO/SRO		2.9	3.4
10CFR Link		(CFR: 41.10 / 45.13)	

# 20	☐ RO	☑ SRO	Question ID: 3100015	Origin: New	☐ Memory? (Check=Yes)

The Plant has tripped from 100% power with the following complications:

- Two (2) CEAs have stuck fully withdrawn.
- VA-10 was lost at the time of the trip.
- The Charging Header has ruptured resulting in a small-break LOCA.
- An Excess Steam Demand has occurred on the #1 SG.

EOP-2540 has been entered, actions have been taken to isolate feed to the #1 SG and it has completed blowing down. RCS temperature has stabilized at approximately 420°F.

Which one of the following statements dealing with Technical Specification requirements applies in the existing situation?

A Shutdown Margin can NOT be considered met until the RCS boron concentration has been raised to account for both of the stuck CEAs. ☑

B The Charging System can NOT be considered OPERABLE until the Alternate Charging Path has been fully established. ☐

C ESAS can NOT be considered OPERABLE until ALL Facility One ESAS components have been manually actuated or aligned to their accident position. ☐

D A manual plant cooldown can NOT be started until the RCS has soaked for that time required to meet the RCS Cooldown Limits from 100% power temperatures. ☐

Justification A - CORRECT; The shutdown Margin curves take into account the most reactive CEA has stuck out, but ONLY one CEA. If more than one CEA has stuck out on a trip, RCS boron concentration must be raised to account for the extra CEA, thereby ensuring SDM is still met.
B - WRONG; The Alternate Charging Path can NOT be used to take credit as an OPERABLE Charging Path.
C - WRONG; Facility 1 of ESAS is INOPERABLE because of the loss of VA-10. Manual alignment of equipment does not make it operable, it just performs the funtion in its place.
D - WRONG; The temperature used for the Cooldown Limit is "reset" once the RCS cooldown from the ruptured SG is stopped. Therefore, a plant cooldown should continue from 420°F, not to exceed the TS limit from that point on.

Reference MP2*LOUT, 2541, Appnd. 17, TS Bases, TRM

NRC K/A System/E/A		NRC K/A Generic
System	E09 Functional Recovery	Conduct of Operations
Number	GA	2.1.33
	SEE GENERIC K/A	Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.
Importance RO/SRO		3.4 4.0
10CFR Link		(CFR: 43.2 / 43.3 / 45.3)

# 21	☐ RO ☑ SRO	Question ID: 3100007	Origin: New	☐ Memory? (Check=Yes)

The plant is operating normally at 100% power. I&C personnel have removed the #1 'alternate' steam flow detector for a scheduled calibration. The #1 steam flow selector switch was placed in the 'MAIN' position on C-05 prior to the start of the calibration.

Approximately 15 minutes after the start of the calibration, the SPO reports the following:
- SG LEVEL SET POINT DEVEATION HI/LO annunciator
- #1 and #2 SG levels are lowering.
- #1 SG steam flow indication on the SGTR screen is lowering and appears to be caused by a failing detector.

Which of the following actions will the US direct to stabilize the plant?

A Take manual control of #1 FRV. Place BOTH SGFP speed controls in 'MANUAL' and maintain FRV D/P to between 20 and 100 psid while manually restoring #1 SG level to 70%. ☑

B Take 'MANUAL' control of BOTH SG FRVs and position them to restore SG levels to the normal level band. Ensure the SGFPs automatically maintain the required delta-P. ☐

C Take 'MANUAL' control of the #1 FRV and the "A" SGFP. Restore #1 SG level to 70% while operating the "A" SGFP in manual to maintain the required delta-P. ☐

D Take 'MANUAL' control of BOTH SG FRVs and position them to restore SG levels. Open the CPF bypass valve to raise SGFP suction and discharge pressure and compensate for the greater demand in feed flow. ☐

Justification A is correct. The steam flow detector is an input to automatic control of SGFP speed as well as automatic FRV position. ARP 2590D requires manual control of SGFP speeds and manual control of #1 FRV on a failure of the steam flow detector.
B is incorrect. Because there is nothing wrong with the #2 SG FRV control system, placing it in manual adds an unnecessary burden to the recovery of the plant. NOT placing both SGFPs in manual will prevent operator mitigation of this event.
C is incorrect. BOTH SGFPs need to be manually controlled to mitigate this failure. Controlling only one will allow the other to shut down, and one SGFP cannot supply sufficient feed flow above 75% power.
D is incorrect. Controlling #2 SG FRV in manual is counter-productive. Opening the CPF bypass will raise SGFP suction pressure, but the control system will then LOWER SGFP speed even faster in an attempt to maintain a perceived lower need of delta-P.

Reference MP2*LOUT, MFW, 2321, ARP 2590D, MB-06184

NRC K/A System/E/A		NRC K/A Generic
System	059 Main Feedwater (MFW) System	Emergency Procedures /Plan
Number	GS	2.4.49
	SEE GENERIC K/A	Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.
Importance RO/SRO		4.0 4.0
10CFR Link		(CFR: 41.10 / 43.2 / 45.6)

# 22	☐ RO ☑ SRO	Question ID: 3155356	Origin: Modified	☐ Memory? (Check=Yes)

The plant is in a refueling outage with fuel movement in progress. The Refueling Machine operator retrieved a bundle from the appropriate upender location and has just indexed the refuel machine over the appropriate core location, B-5, with a mast rotation of 90 degrees in preparation for lowering a new fuel bundle into the core.

As the Refueling SRO, what action must you perform and why?

A Direct the operator to move the refuel machine to a Core Clear Area, rotate the mast with the assembly to the '180' degree position, return the bundle to the correct core location, and lower the bundle into the core. ☐
Rotating the mast will ensure the mast is oriented as required by the core map and will prevent damage to the camera.

B Stop fuel movement. Request the Shift Manager to determine and approve an appropriate mast rotation and enter this information in the Shift Manager's log prior to resuming fuel movement. ☐
Rotating the mast will ensure the mast is oriented as required by the core map and will prevent damage to the camera.

C Stop fuel movement. Request Reactor Engineering to determine and approve the appropriate course of action and modify the Materials Transfer Form prior to resuming fuel movement. ☑
The camera must be rotated to prevent damage, but rotating the bundle could result in an unexpected radial flux pattern.

D Direct the operator to return the bundle to the upender, ungrapple and rotate the mast to the '180' degree position, return the bundle to the correct core location, and lower the bundle into the core. ☐
The camera must be rotated to prevent damage, but rotating the bundle could result in an unexpected radial flux pattern.

Justification C is correct. Although the fuel bundle is in the correct orientation, the mast is NOT. (Lowering the bundle with the mast at 90 degrees will result in damaging the camera and likely cause parts of the camera to fall into the vessel.) Fuel movement must be stopped to allow Reactor Engineering to modify the Materials Transfer Form before fuel movement can resume. All moves must be performed in accordance with the Materials Transfer Form. The Refuel SRO cannot deviate from this document. An improperly rotated assembly may result in an unexpectred radial flux pattern depending on several factors such as core location, number of times burned, proximity to detectors, etc.
A is incorrect. Although the camera may be damaged if the mast is NOT rotated, the Refuel SRO does NOT have the authority to deviate from the Materials Transfer Form by rotating the mast and placing the fuel bundle in an unauthorized position.
B is incorrect. Although the camera may be damaged if the mast is NOT rotated, the Shift Manager does NOT have the authority to deviate from the Materials Transfer Form by moving the fuel bundle in an unauthorized position.
D is incorrect. Although this is ultimately the correct course of action, the Refuel SRO does NOT have the authority to direct a deviation from the Materials Transfer Form. Reactor Engineering must make the appropriate changes to the form and have them approved prior to making any fuel movement.
Requires the use of the core map

Reference MP2*LOUT, 2303, Refuel, MB-04841
Requires the use of the core map

NRC K/A System/E/A

System 034 Fuel Handling Equipment System (FHES)

Number A2.03

Ability to (a) predict the impacts of the following malfunctions or operations on the Fuel Handling System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Mispositioned fuel element

Importance RO/SRO 3.3 4.0

10CFR Link (CFR: 41.5 / 43.5 / 45.3 / 45.13)

NRC K/A Generic

# 23	☐ RO	☑ SRO	Question ID: 3171665	Origin: Bank	☐ Memory? (Check=Yes)

The US will direct entry into AOP 2515, Loss of Containment Integrity, when _____ and the plant is in ____.

A The inner containment airlock door has passed its leak test and is closed and locked, but outer containment airlock door fails its leak test; MODE 1 ☐

B One automatic containment isolation valve exceeds its allowable stroke time; MODE 4 ☑

C Containment leak rate exceeds its maximum Technical Specification requirement; MODE 5 ☐

D The containment purge system supply and exhaust valves are open with both containment airlock doors open; MODE 6 (during fuel movement) ☐

Justification B is correct. The entry conditions for AOP 2515 includes: "the unit is in MODES 1, 2, 3, or 4 and an automatic containment isolation valve fails to position to the required accident position during surveillance OR fails to stroke properly when operated."
A is incorrect. The entry conditions for AOP 2515 includes: "Both doors of the containment airlock are open OR have indications of a failed leak test." In this case, only one airlock door has failed its leak test. The inner door would be closed.
C is incorrect. This procedure is NOT applicable in MODE 5.
D is incorrect. This is an allowable configuration in MODE 6 (during fuel movement).

Reference MP2*LOUT AOP 2515, CTMT, MB-05558

NRC K/A System/E/A		NRC K/A Generic
System	103 Containment System	Emergency Procedures /Plan
Number	GS	2.4.4
	SEE GENERIC K/A	Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.
Importance RO/SRO		4.0 4.3
10CFR Link		(CFR: 41.10 / 43.2 / 45.6)

# 24	☐ RO ☑ SRO	Question ID: 3100016	Origin: New	☐ Memory? (Check=Yes)

The Plant is at 100% power when the "A" Safety Channel of Steam Generator Pressure, PT-1023A, fails low due to a transmitter failure. All actions required by plant Technical Specifications are then carried out by the Crew.

Which one of the following describes why Appendix "R" administrative requirements do NOT dictate further actions for the loss of the "A" SG pressure safety instrument?

A Safety Channel "A" is not a Facility Two instrument. ONLY Facility two instruments are qualified to meet the Appendix "R" bases for availability during an Appendix "R" fire. ☐

B Safety Channel "A" of SG pressure, although a Safety instrument, is not qualified as an Appendix "R" instrument and does not feed any Appendix "R" panels or equipment. ☑

C All Channel "A" Safety instrumentation is powered from VA-10, which is intentionally deenergized in all Appendix "R" fires to prevent a hot short from inadvertently actuating equipment. ☐

D The SG Safety range pressure transmitters do not have the range for a plant shutdown and subsequent cooldown, which is the assumed operational progression in all Appendix "R" fires. ☐

Justification A - WRONG; Tech. Spec. facility designation is not a criteria used for Appendix "R" inclusion, even though the C-10 Fire Shutdown Panel is Facility Two specific. The Appendix "R" bases lists states that Tech. Spec. facility operability requirements do not apply to App. "R" components.
B - CORRECT; App. "R" specifications listed in the TRM do not include all Tech. Spec. equipment. App. "R" bases states inclusion in this criteria is not based on Tech. Spec. requirements.
C - WRONG; VA-10 is not intentionally deenergized for an App. "R" fire, but the App. "R" concerns are met by local isolation of specific equipment of concern. VA-10 is intentionally deenergized for a Station Blackout situation and control power at C-10 is powered by VA-20.
D - WRONG; The range of this instrument is 0 - 1000 psia, it is the safety channel level instrumentation that has a narrow range for normal operation.

Reference MP2*LOUT, Appnd. "R", TRM Bases

NRC K/A System/E/A		NRC K/A Generic
System	067 Plant fire on site	Equipment Control
Number	GA	2.2.25
	SEE GENERIC K/A	Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.
Importance RO/SRO		2.5 3.7
10CFR Link		(CFR: 43.2)

# 25	☐ RO ☑ SRO	Question ID: 3100000	Origin: New	☐ Memory? (Check=Yes)

With the plant operating at 100% power with Bus 24E aligned to Bus 24C, the following alarms are received:

- RBCCW HDR B PRESS LO (C-06/7)
- RBCCW HDR B FLOW HI (C-06/7)
- RBCCW SURGE TK LEVEL HI/LO (C-06/7)
- AUX BLDG SUMP LEVEL HI (C-06/7)
- PMW HEADER LOW PRESSURE (C-02/3)
- Various low flow annunciators for components supplied by "B" RBCCW header

Which of the following actions will the US direct and why?

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

B Isolate the "C" RBCCW Heat Exchanger, secure RBCCW Surge Tank make up, trip the Reactor, secure the "B" and ""D RCPs, and perform Standard Post Trip Action, EOP 2525. The "B" RBCCW Header has ruptured on the outlet nozzle of the "C" RBCCW Heat Exchanger. ☐

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

D Place the "C" RBCCW Pump in Pull-To-Lock secure RBCCW Surge Tank make up, trip the Reactor, secure the "B" and ""D RCPs, and perform Standard Post Trip Action, EOP 2525. The "B" RBCCW Header has ruptured on the piping that supplies the Letdown Heat Exchanger, Sample Coolers, and the Degassifier. ☑

Justification D is correct. A header rupture on the piping that supplies the Letdown Heat Exchanger, Sample Coolers, and the Degassifier will indicate high flow on the "B" RBCCW header flow instrument. Because the rupture is unisolable, the "B" Pump must be placed in Pull-To-Lock to prevent pump out more fluid and the Reactor must be tripped (along with the "B" and "D" RCPS) due to loss of cooling to "B" and "D" RCPs.
"A" is incorrect but credible because the student may mistakenly believe that a rupture has occurred on the inlet to the "C" RBCCW Heat Exchanger that can be isolated by removing the "C" RBCCW Pump and Heat Exchanger from service. Additionally, the Sump Level alarm is indicative of a rupture in a different location.
"B" is incorrect but credible because the student may mistakenly believe that the rupture is on the outlet of the "C" RBCCW Heat Exchanger. Tripping the plant may be acceptable; however, the Aux Building Sump Level alarm and high header flow are indicative of a rupture in a different location such that isolation of the heat exchanger would NOT isolate the leak.
"C" is incorrect but credible because the student may mistakenly believe that the flow instrument would indicate a high flow if the RBCCW Surge Tank supply line ruptured. Additionally, a rupture in this location could NOT simultaneously indicate a high flow and a low header pressure.

Reference MP2*LOUT, RBCCW, 2564, MB-04726

NRC K/A System/E/A

System 026 Loss of Component Cooling Water (CCW)

Number AA2.02

Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The cause of possible CCW loss

Importance RO/SRO 2.9 3.6

10CFR Link (CFR: 43.5 / 45.13)

NRC K/A Generic