DRAFT HARRIS RO Q. 1-75

2011 HNP SRO NRC Written Exam

1. 2011 NRC RO 001/NEW/FUNDAMENTAL////EARLY SUBMITTAL/007 EK1.05/

Given the following plant conditions:

- The plant was operating at 100% for 30 days and has just tripped
- The crew is implementing EPP-004, Reactor Trip Response
- The SRO is directing the OAC to check Source Range detector status

(1) What flux level will the Intermediate Range detectors indicate when the Source Range instruments energize?

(2) How long after the Reactor trip will it be before the Intermediate Range detectors reach this level?

A.✓ (1) 5×10^{-11} Amps

(2) <15 minutes

B. (1) 5×10^{-11} Amps

(2) >20 minutes

C. (1) 1×10^{-10} Amps

(2) <15 minutes

D. (1) 1×10^{-10} Amps

(2) >20 minutes

Feedback

Plausibility and Answer Analysis

IAW EPP-004 the Intermediate Range amp will be $\leq 5 \times 10^{-11}$Amps when the Source Range instruments energize. *The time is also correct based on a negative 1/3 decade per minute startup rate after a Reactor trip (Actual recorded plant data after a trip was 9 minutes). Using the equation* $P = P_0 SUR^t$ *where* P_0 *is 0% power intermediate range equivalent and SUR is -1/3 DPM based on longest lived delayed neutron precursors, t = 18 minutes. Therefore when the short lived precursors are considered, the time can be determined to be less than 18 minutes.*

A. Correct.

B. Incorrect. Plausible because the Intermediate Range amp reading is correct but the time to reach this level is incorrect. The > 20 minutes is the approximate time that the RWST will be depeleted during a LB LOCA.

C. Incorrect. Plausible because during a Reactor Start-up the Source Range Detectors are de-energized at 1×10^{-10} *Amps (P-6 setpoint). The time to reach this level is correct.*

D. Incorrect. Plausible because during a Reactor Start-up the Source Range Detectors are de-energized at 1×10^{-10} *Amps (P-6 setpoint).*
The > 20 minutes is the approximate time that the RWST will be depeleted during a LB LOCA.

Notes

000007 (BW/E02&E10; CE/E02) Reactor Trip - Stabilization - Recovery / 1

007EK1.05 Knowledge of the operational implications of the following concepts as they apply to the reactor trip: Decay power as a function of time
(CFR 41.8 / 41.10 / 45.3)

Importance Rating:	3.3 3.8
Technical Reference:	EOP-EPP-004, Step 18, Page 32 Rev. 21
References to be provided:	None
Learning Objective:	EOP-LP-3.1 Objective 3.a
Question Origin:	NEW
Comments:	None
Tier/Group:	T1G1

2. 2011 NRC RO 002/NEW/FUNDAMENTAL/////009 EG2.4.20/

Given the following plant conditions:

- The plant was operating at 100% power
- The crew performed a manual Reactor Trip and Safety Injection due to RCS leakage in excess of makeup capability
- They have secured one CSIP and have just completed realigning the CSIP discharge from the BIT to the normal Charging line per EPP-008, SI Termination

The CRS has directed the OAC to maintain PZR level

Which ONE of the following completes the statement below?

Charging flow must NOT exceed __(1)__ gpm to prevent __(2)__ IAW the EPP-008 caution.

A. (1) 120

(2) exceeding normal RCS makeup capability

B. (1) 120

(2) damage to the Regenerative Heat Exchanger

C. (1) 150

(2) exceeding normal RCS makeup capability

D✓ (1) 150

(2) damage to the Regenerative Heat Exchanger

Feedback

Plausibility and Answer Analysis

The caution in EPP-008 prior to controlling Charging flow to maintain PRZ level warns the operator that Charging flow should NOT exceed 150 gpm to prevent damage to the regenerative heat exchanger.

A. Incorrect. *Plausible because 120 gpm is the maximum normal makeup capability.*

B. Incorrect. *Plausible because 120 gpm is the is the maximum normal makeup capability and the second part of the answer is correct.*

C. Incorrect. *Plausible because the 150 gpm flow rate is correct but the reason for the maximum flow rate is incorrect.*

D. Correct.

Notes

000009 Small Break LOCA / 3

009EG2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes. (CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	3.8 4.3
Technical Reference:	EPP-008, Rev. 24, Caution prior to step 12 and CVCS Student text
References to be provided:	None
Learning Objective:	EOP-LP-3.1 Objective 5
Question Origin:	NEW
Comments:	Meets KA by asking the candidate about a caution in the Small Break LOCA procedure dealing with the maximum flow rate of a system to prevent damage to system components.
Tier/Group:	T1G1

3. 2011 NRC RO 003/NEW/C/A/////011 EK2.02/

Given the following plant conditions:
- The plant was operating at 100% power
- A LOCA occurred

Current conditions are:
- The crew has just completed checking Main Steam Isolation per step 9 of PATH-1
- The 'A' and 'B' CSIP will not start
- All RCPs are operating
- Containment pressure is 11.3 psig and rising
- RCS pressure is 1050 psig
- 'A' Containment Spray pump has tripped, 'B' Containment Spray pump is running
- Pressurizer level is offscale low
- RCS subcooling is 0°F

Which ONE of the following describes the crew response to this situation?

A.✓ Trip all RCPs and continue with PATH-1

B. Leave all RCPs running and continue with PATH-1

C. Trip all RCPs and immediately transition to FRP-J.1, Response to High Containment Pressure

D. Leave all RCPs running and immediately transition to FRP-J.1, Response to High Containment Pressure

Feedback

Plausibility and Answer Analysis

PATH-1 fold out page for RCP trip criteria has not been met (SI flow > 200 gpm and RCS pressure < 1400 psig) due to failure of the Charging pumps; however, with >10 psig in containment phase B actuates and isolates CCW to the RCPs. PATH-1 step 10 checks Containment pressure < 10 psig. If pressure was > 10 psig the operator is directed to stop all RCPs. RCPs are not run without support conditions unless the core cooling is degraded to the point of entering FRP-C.1 at >1200°F CETC.

A. Correct.

B. Incorrect. Plausible because RCP trip criteria has two parts, RCS pressure < 1400 psig and SI flow > 200 gpm. With both CSIPs not running SI flow would be 0 gpm and the candidate could think that RCPs should be left running with this condition. With Containment pressure at 11.3 psig a Phase B will occur (>10 psig). RCPs are not run without support conditions unless the core cooling is degraded to the point of entering FRP-C.1 at >1200°F CETC.

C. Incorrect. First part is correct. Second part is plausible because a tranistion to FRP-J.1 (Containment CSF) would be made if Containment Pressure is > 10 psig and less than 2 Containment Spray pumps were running. Even without Containment Spray pumps in operation Critical Safety Function Status Trees are not implemented until later in PATH-1.

D. Incorrect. First part is plausible because RCP trip criteria has two parts, RCS pressure < 1400 psig and SI flow > 200 gpm. With both CSIPs not running SI flow would be 0 gpm and the candidate could think that RCPs should be left running with this condition. Second part is plausible because a tranistion to FRP-J.1 (Containment CSF) would be made if Containment Pressure is > 10 psig and less than 2 Containment Spray pumps were running. Even without Containment Spray pumps in operation Critical Safety Function Status Trees are not monitored until later in PATH-1.

Notes

000011 Large Break LOCA / 3

011EK2.02 Knowledge of the interrelations between the and the following Large Break LOCA: Pumps
(CFR 41.7 / 45.7)

Importance Rating:	2.6 2.7
Technical Reference:	PATH-1 Guide, Step 10, Rev. 30
References to be provided:	None
Learning Objective:	EOP-LP-3.1 Objective 3.c
Question Origin:	NEW
Comments:	Meets K/A by questioning the relationship between a Large Break LOCA and the operation of the RCPs
Tier/Group:	T1G1

4. 2011 NRC RO 004/NEW/C/A/////015 AK3.03/

Given the following plant conditions:

- The plant is operating at 55% power
- Annunciator ALB-010-1-5, RCP-B TROUBLE has alarmed
- The crew is implementing AOP-018, Reactor Coolant Pump Abnormal Conditions
- The OAC identifies that the red light on the 'B' RCP shaft vibration monitor is lit and frame vibrations are increasing

Which ONE of the following completes the statements below?

IAW AOP-018, the 'B' RCP has reached its trip limit when shaft vibration levels exceed __(1)__ mils. The reason that the Reactor would be tripped prior to tripping the affected RCP is __(2)__ .

A. (1) 5

(2) the plant will be in an unanalyzed condition if the RCP is tripped with Reactor power greater than P-8.

B. (1) 5

(2) operators are expected to take manual actions prior to reaching the automatic setpoint for prescribed RPS actuations.

C. (1) 20

(2) the plant will be in an unanalyzed condition if the RCP is tripped with Reactor power greater than P-8.

D✓ (1) 20

(2) operators are expected to take manual actions prior to reaching the automatic setpoint for prescribed RPS actuations.

Feedback

Plausibility and Answer Analysis

AOP-018 RCP trip setpoint for high vibration on a RCP shaft is 20 mils. The RCP high vibrations could indicate a loose impleller or a damaged or loose bearing, while high frame vibration levels may indicate coupling misalighment or an unbalanced shaft. There is no automatic RCP trip on high vibrations therefore the RCP must be manually tripped. If an RCP is tripped while above the P-8 setpoint (≥49% power) an automatic Reactor trip will be generated.

IAW OPS-NGGC-1000 section 4.6 for RO responsibilities - the ROs may actuate Emergency Safeguards Features whenever the corresponding setpoints and logic are met or will be met. When manually actuating ESF, RPS, or inserting a manual scram, the ROs shall inform the CRS about the action to be taken. If time permits, this notification should be made prior to taking the action. This communication should not delay action when safety of the reactor or plant is in jeopardy.

A. Incorrect. *Plausible because RCP trip criteria for high vibrations on the frame is 5 mils. The second part is plausible because if a Reactor trip did not occur then the plant could be in an unanalyzed condition but above 49% (P-8) a trip of a single RCP will cause an Automatic Reactor trip. The Reactor Trip condition will not be an unanalyzed condition.*

B. Incorrect. *Plausible because RCP trip criteria for high vibrations on the frame is 5 mils. The second part of answer is correct.*

C. Incorrect. *Plausible because 20 mils shaft vibration is correct. The second part of the answer is plausible because if a Reactor did not occur then the plant could be in an unanalyzed condition but above 49% (P-8) and a trip of a single RCP will cause an Automatic Reactor trip. The Reactor Trip condition will not be an unanalyzed condition.*

D. Correct.

Notes

000015/17 RCP Malfunctions / 4

015AK3.03 Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow) :Sequence of events for manually tripping reactor and RCP as a result of an RCP malfunction
(CFR 41.5,41.10 / 45.6 / 45.13)

Importance Rating:	3.7 4.0
Technical Reference:	AOP-018 Section 3.1 steps 2-8 and Attachment 1 step 5 RCP vibration trip limits, Rev 39, and OPS-NGGC-1000 RO responsibilities Section 4.6.3 Page 27, Rev 3
References to be provided:	None
Learning Objective:	AOP-LP-3.18 Objective 3.a
Question Origin:	NEW
Comments:	None
Tier/Group:	T1G1

5. 2011 NRC RO 005/BANK/C/A/ANSWER SWITCHED////022 AA2.03/

Given the following plant conditions:

- The plant is operating at 100% power
- Rod Control is in Manual
- Automatic makeup to the VCT secured approximately 10 minutes ago and the system is in AUTO
- The OAC notices that T_{avg} started to decrease

Which ONE of the following identifies the cause of the decrease in RCS temperature?

A.✓ FCV-114B, REACTOR MAKEUP WATER TO BORIC ACID BLENDER, had a loss of air to the valve during the Auto Makeup.

B. FK-113, BORIC ACID MKUP FLOW, controller has failed low during the Auto Makeup.

C. A newly replaced CVCS mixed bed demineralizer was put in service.

D. The mixed bed demineralizer is exhausted.

Feedback

Plausibility and Answer Analysis

FCV-114B fails closed on a loss of air. With this valve closed during the auto MU, there would be only a boration flow path and therefore would borate the RCS and cause Tavg to decrease since there would be no RMW flow.

A. Correct

B. Incorrect — *Plausible if the candidate confuses the results of this failed controller. With this controller failed low, FCV-113A would go closed or remain closed and this would cause a dilution event and the RCS temperature would increase.*

C. Incorrect — *Plausible if the candidate confuses the results of a placing a new CVCS mixed bed demineralizer in service. The result would dilute the RCS causing Tavg to increase not decrease.*

D. Incorrect — *Plausible if the candidate confuses the results of a depleted mixed bed demineralizer. Incorrect because a depleted mixed bed would not change reactivity and therefore not cause an RCS temperature change.*

Notes

000022 Loss of Rx Coolant Makeup / 2

022AA2.03 Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: Failures of flow control valve or controller
(CFR 43.5/ 45.13)

Importance Rating:	3.1 3.6
Technical Reference:	AOP-017, Attachment 1, Page 40 Rev. 32
References to be provided:	None
Learning Objective:	AOP-LP-3.17 Objective 3.a.1
Question Origin:	Bank
Comments:	None
Tier/Group:	T1G1

6. 2011 NRC RO 006/NEW/C/A/////025 AA2.01/

Given the following plant conditions:

- The plant is in Mode 5
- RHR Train 'A' is in service in shutdown cooling mode
- FK-605A1, RHR Heat Exchanger 'A' Bypass Flow Controller (1RH-20), is in AUTO
- A rupture has occurred in RHR piping just upstream of FT-605A

Compared to prior to the rupture, A-SA RHR current indication will ____(1)____ and FK-605A1 output will ____(2)____ .

A. (1) decrease (2) decrease

B. (1) decrease (2) increase

C. (1) increase (2) decrease

D✓ (1) increase (2) increase

Feedback

Plausibility and Answer Analysis

Pump current will increase due to the flow increase from the pipe rupture and 1RH-20 openning further since the actual flow through FT-605A will decrease causing the flow controller to demand more flow by openning 1RH-20.

A. Incorrect. The second part is plausible since the affect on FT-0605A for a leak downstream of the FT would decrease the output and then correctly applying the relationship between pump flow and current make the first part plausible.

B. Incorrect. The second part is correct. The first part is plausible since the applicant may incorrectly understand system configuration or mis-apply the relationship between pump flow and current.

C. Incorrect. The first part is correct. The second part is plausible since the affect on FT-0605A for a leak downstream of the FT would decrease the output.

D. Correct.

Notes

000025 Loss of RHR System / 4

025 AA2.01 Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Proper amperage of running LPI/decay heat removal/RHR pump(s)
(CFR: 43.5 / 45.13)

Importance Rating:	2.7 2.9
Technical Reference:	GFES - No reference provided
References to be provided:	None
Learning Objective:	RHR Objective 8
Question Origin:	NEW
Comments:	None
Tier/Group:	T1G1

7. 2011 NRC RO 007/BANK/C/A/ANSWER SWITCHED////026 AA1.06/
Given the following plant conditions:
- The plant is operating at 100% power
- The temperature input to TK-144, LTDN Temperature (1CC-337) fails low

Which ONE of the following describes the impact on the plant and the action that is required by the appropriate Annunciator Panel Procedure (APP)?

A.✓ High temperature letdown diversion to VCT;

Place TK-144 in manual and adjust flow to restore normal cooling.

B. High temperature letdown diversion to VCT;

Isolate letdown and place Excess Letdown in service.

C. Low Letdown temperature causing Tavg to rise.

Place TK-144 in manual and adjust flow to restore normal cooling.

D. Low Letdown temperature causing Tavg to rise.

Isolate letdown and place Excess Letdown in service.

Feedback

Plausibility and Answer Analysis

The input to TCV-144 failing low will cause CCW to throttle flow down to the HX in an attempt to raise temperature. This will cause actual temperature to increase and an automatic diversion to the VCT will occur to protect the demineralizers. Correct response is to take manual control of TCV-144 and restore proper cooling flow.

A. Correct.

B. Incorrect. Plausible because this would be performed if after placing the TCV in manual and the valve still did not respond correclty.

C. Inorrect. Plausible if a low temperature failure would cause the CCW flow control valve to open, lowering Letdown temperature, which will cause more boron to be removed in the demins resulting in positive reactivity (dilution) which will cause Tavg to rise. The second part is correct for the failure.

D. Incorrect. Plausible if a low temperature failure would cause the CCW flow control valve to open, lowering Letdown temperature, which will cause more boron to be removed in the demins resulting in positive reactivity (dilution) which will cause Tavg to rise. The second part is plausible because if the Letdown Heat Exchanger was not functioning correclty then letdown would be isolated and excess letdown would be placed in service.

Notes

000026 Loss of Component Cooling Water / 8

026AA1.06 Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: Control of flow rates to components cooled by the CCWS

Importance Rating:	2.9 2.9
Technical Reference:	ALB-007-3-2 page 13 Rev. 14
References to be provided:	None
Learning Objective:	AOP-LP-3.14 Objective 3
Question Origin:	Bank OIT Dev 008 A2.08 1
Comments:	None
Tier/Group:	T1G1

8. 2011 NRC RO 008/BANK/FUNDAMENTAL/////027 AK1.01/
With the plant operating at 100% power which ONE of the following PZR Pressure Control System Malfunctions will cause PZR Saturation Temperature to RISE?

A. PZR Spray Valve failed open

B.✓ PZR Master Controller output fails low

C. PZR Master Controller setpoint fails low

D. PZR Pressure input to Master Controller fails high

Feedback

Plausibility and Answer Analysis

Anything that will raise pressurizer pressure will raise the saturation temperature of the pressurizer. Answers A, B, and D are all conditions that will cause Pressurizer pressure to decrease.

A. Incorrect. *Plausible because a spray valve failure will introduce cooler water to the PRZ but the liquid temperature decease will not have the same magnitude of affect that the lowering of the PRZ pressure will have on saturation temperature. The lower pressure will cause saturation temperature to lower.*

B. Correct. *The output lowering will cause heaters to turn on to raise pressure.*

C. Incorrect. *Plausible because the candidate could be confused on how the setpoint failure affects PRZ pressure. The PRZ spray valves are controlled by the Master controller. If the pressure must be increased the demand goes down. Therefore someone could be confused that a low failure would cause a pressure increase. But, if the setpoint fails low, the controller will attempt to maintain a lower pressure, which will result in a lower saturation temperature.*

D. Incorrect. *Plausible similar to 'B' where confusion could occur on a high failure of the master controller. In this case the failure would cause the spray valves to close but pressure will actually be reduced due to PORV 444-B opening from the master controller failure. The opening of the PORV will reduce RCS pressure causing saturation temperature to lower. (until RCS pressure reaches 2000 psig at which point the PORV will auto close).*

Notes

000027 Pressurizer Pressure Control System Malfunction / 3

027AK1.01 Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions: Definition of saturation temperature
(CFR 41.8 / 41.10 / 45.3)

Importance Rating: 3.1 3.4

Technical Reference: AOP-019 Rev. 22 Attachment 3

References to be provided: None

Learning Objective: PZRPC Objective 9

Question Origin: Bank

Comments: None

Tier/Group: T1G1

9. 2011 NRC RO 009/NEW/C/A////EARLY SUBMITTAL/029 EK2.06/

Given the following plant conditions:

- Reactor Trip testing is in progress on Train 'A'
- Reactor Trip Breaker 'A' is open
- Reactor Trip Bypass Breaker 'A' is closed
- A transient occurs requiring a Reactor Trip
- The OAC attempts to manually trip the Reactor but the Reactor does NOT trip

Which ONE of the following describes the problem that has caused the failure of the Reactor to trip?

A.✓ Reactor trip Bypass Breaker 'A' Undervoltage trip coil failed to deenergize.

B. Reactor trip Bypass Breaker 'A' Shunt trip coil failed to deenergize.

C. Reactor trip Breaker 'B' Undervoltage trip coil failed to energize.

D. Reactor trip Breaker 'B' Shunt trip coil failed to deenergize.

Feedback

Plausibility and Answer Analysis

The Reactor trip breakers and bypass breakers function is to open on an automatic or manual trip signal, which will interrupt power from the Rod Control system to the CRDMs, causing the rods to fall to the bottom of the core. Reactor trip breakers A and B are normally shut and are in series such that opening either breaker will cause the rods to drop into the core. A bypass Reactor trip breaker is connected in parallel with each reactor trip breaker (bypass A is in parallel with trip breaker A; bypass B is in parallel with trip breaker B). The bypass breakers are normally open and racked out, but may be racked in and shut to allow testing of the associated trip breaker. Each reactor trip and bypass breaker has redundant trip coils: An UV trip coil and a shunt trip coil. The UV coil is maintained energized by the output of the SSPS logic bay when a trip signal is not active. When a reactor trip is initiated by SSPS, power to the UV coil is removed, causing the Reactor trip breaker to open. The shunt trip coil, normally de-energized, is energized by 125 VDC power when a Reactor trip is active.

A. Correct

B. Incorrect. Plausible due to 'B' Reactor trip bypass breaker having a shunt trip coil, however it is energized to trip not deenergize to trip.

C. Incorrect. Plausible due to 'B' Reactor trip breaker having a UV coil, however it is deenergized to trip not energize to trip.

D. Incorrect. Plausible due to 'B' Reactor trip breaker having a shunt trip coil, however it is energized to trip not deenergize to trip.

Notes

000029 ATWS / 1

029EK2.06 Knowledge of the interrelations between the and the following an ATWS: Breakers, relays, and disconnects
(CFR 41.7 / 45.7)

Importance Rating:	2.9* 3.1*
Technical Reference:	Drawings Emdrac 1364-0865 and CWD's 6-b-401 sheet 91 (A train) and sheet 93 (B train).
References to be provided:	None
Learning Objective:	RPS Objective 4.c
Question Origin:	NEW
Comments:	None
Tier/Group:	T1G1

10. 2011 NRC RO 010/BANK/C/A/////038 EA1.36/

Given the following plant conditions:
- PATH-2 is being performed.
- The MSIV on the ruptured SG is mechanically stuck open.
- The Main Steam Isolation Valves (MSIVs) on the intact SGs are closed.
- A cooldown from 557°F to 485°F at the maximum rate is required.

Which ONE of the following describes the method to accomplish this cooldown in accordance with PATH-2 and the EOP User's Guide?

A.✓ Fully open the intact SG PORVs as fast as possible

B. Fully open the Steam Dumps as fast as possible without causing a Main Steam Line Isolation signal

C. Fully open the Steam Dumps as fast as possible

D. Fully open the intact SG PORVs as fast as possible without causing a Main Steam Line Isolation signal

Feedback

Plausibility and Answer Analysis

During a SGTR cooldown only the intact SGs should be used to cooldown the RCS and since the MSIVs on the intact SGs are closed, the PORVs should be used. The valves should be opened as fast as possible since generation of an MSLI signal is not a concern.

A. Correct.

B. Incorrect. *Plausible since the maximum cooldowrn rate is desirable using maximum steam dump flow without causing too great a rate of pressure drop will result in the MSIVs going closed, but it is also undesirable to use steam dumps when the ruptured SG MSIV is open.*

C. Incorrect. *Plausible since the maximum cooldown rate can be achieved using maximum steam dump flow, but causing too great a rate of pressure drop will result in the MSIVs going closed which is undesirable and it is also undesirable to use steam dumps when the ruptured SG MSIV is open.*

D. Incorrect. *Plausible since causing the MSIVs to close is not desirable when steam dumps are being used, but when already using PORVs to dump steam this is not a concern.*

Notes

000038 Steam Generator Tube Rupture (SGTR) / 3

038EA1.36 Ability to operate and monitor the following as they apply to a SGTR:
Cooldown of RCS to specified temperature
(CFR 41.7 / 45.5 / 45.6)

Importance Rating:	4.3 4.5
Technical Reference:	PATH-2 Step 11 Rev. 23
References to be provided:	None
Learning Objective:	EOP-LP-3.2 Objective 1.c
Question Origin:	Bank
Comments:	None
Tier/Group:	T1G1

11. 2011 NRC RO 011/NEW/C/A////WE12 EK1.3/

Given the following plant conditions:

- The plant was operating at 100% power when a Main Steam line break occurred
- A Reactor Trip and Safety Injection has been actuated
- The crew is implementing EPP-015, Uncontrolled Depressurization of All Steam Generators
- Attempts to close the MSIVs have failed
- Safety Injection has NOT been reset

Which ONE of the following:

(1) provides indication in the MCR that a MSLI signal has been generated
AND
(2) if a MSIV can be closed, what plant parameter is monitored to determine when EPP-015 can be exited?

A. (1) A first out annunciator exists for MSLI actuation

(2) RCS loop T-hots

B. (1) Lights are available on the Bypass Permissive Panel for MSLI actuation

(2) RCS loop T-hots

C✓ (1) A first out annunciator exists for MSLI actuation

(2) S/G Pressure

D. (1) Lights are available on the Bypass Permissive Panel for MSLI actuation

(2) S/G Pressure

Feedback

Plausibility and Answer Analysis

ALB-011-2-2 Reactor Trip Stm Ln Isol and SI will annunciate whenever the Low Steam Line pressure (601 psig rate compensated) coincidence is made up. Per EPP-015 foldout criteria a transistion to EPP-014, Faulted SG Isolation, shall be made if any SG pressure increases at any time.

A. Incorrect. Plausible because RCS loop T-hots are monitored for other reasons in this procedure and they will increase once the MSIV is closed but EPP-015 specifies S/G pressure.

B. Incorrect. Plausible because lights exist on the Bypass Permissive Panel for blocks of the MSLI signal and because RCS loop T-hots are monitored for other reasons in this procedure and they will increase once the MSIV is closed but EPP-015 specifies S/G pressure.

C. Correct.

D. Incorrect. Plausible because lights exist on the Bypass Permissive Panel for blocks of the MSLI signal.

Notes

000040 (BW/E05; CE/E05; W/E12) Steam Line Rupture - Excessive Heat Transfer / 4

WE12EK1.3 Annunciators and conditions indicating signals, and remedial actions associated with the (Uncontrolled Depressurization of all Steam Generators).

Importance Rating:	3.4 3.7
Technical Reference:	APP-ALB-011-2-2 Page 5 Rev 7, EPP-015 Foldout Transition to EPP-014 Page 3 Rev 22
References to be provided:	None
Learning Objective:	ESFAS Objective 4, EOP-LP-3.9 Objective 3
Question Origin:	NEW
Comments:	None
Tier/Group:	T1G1

12. 2011 NRC RO 012/BANK/C/A/////054 AA1.01/

Given the following plant conditions:

- The plant was operating at 100% power
- A Reactor trip and Safety Injection have occurred
- The crew has just reached Entry Point C of PATH-1

Current conditions are:

- RCS pressure	1155 psig
- Core Exist T/Cs	564°F
- SG A NR level	42%
- SG B and C NR level	30%
- AFW flow	60 KPPH to each SG
- Containment pressure	3.5 psig
- RWST level	90%
- CST level	7%
- RVLIS Full Range	47%

Which ONE of the following actions is required to be performed next?

A. Implement FRP-H.1, Response To Loss of Secondary Heat Sink

B. Implement FRP-C.2, Response To Degraded Core Cooling

C. Maintain seal injection flow between 8 and 13 gpm

D✓ Align ESW to the AFW pumps

Feedback

Plausibility and Answer Analysis

If CST level is < 10% then the PATH-1 foldout for ESW alignment to the AFW pumps is required.

A. Incorrect. Plausible since total feed flow to the SGs are below the required 210 KPPH but the SG NR levels are above the required transistion requirements (NR level in all SGs are > 25%)

B. Incorrect. Plausible due to the lack of subcooling indicating a degraded core cooling situation. However there is adequate inventory as indicated by RVLIS dynamic range, therefore the Core cooling CSFST remains yellow and this would direct the operator to procedure FRP-C.3 which is not required to be implemented at this time. Also, this is the first direction reached to implement FRP's.

C. Incorrect. Plausible because this is the first operator action after entering PATH-1 entry point C if the operator does not recognize that the foldout criteria for AFW supply switchover is met.

D. Correct

Notes

000054 Loss of Main Feedwater (MFW) / 4

054AA1.01 Ability to operate and / or monitor the following as they apply to the Loss of Main Feedwater (MFW): AFW controls, including the use of alternate AFW sources
(CFR 41.7 / 45.5 / 45.6)

Importance Rating:	4.5 4.4
Technical Reference:	PATH-1 guide Rev 30 Foldout A
References to be provided:	None
Learning Objective:	EOP-LP-3.1 Objective 4
Question Origin:	Bank LORNRC B05 20
Comments:	None
Tier/Group:	T1G1

13. 2011 NRC RO 013/BANK/FUNDAMENTAL/////062 AK3.03/

Given the following plant conditions:

- A LOCA has occurred
- 'A' ESW Booster pump has tripped
- The Crew is implementing FRP-J.1, Response to High Containment Pressure
- Containment pressure is 28 psig

Which ONE of the following identifies why ESW to the 'A' Train Containment Fan Coolers is isolated?

A.✓ To prevent an unmonitored release from Containment to the ESW system.

B. To prevent damage to the containment fan coolers from water hammer due to ESW flashing to steam in piping inside Containment due to low fan cooler flow.

C. To prevent damage to the Containment fan coolers from water hammer if the ESW Booster pump is restarted.

D. To prevent infusion of hydrogen into the ESW system from the Containment atmosphere.

Feedback

Plausibility and Answer Analysis

The ESW Booster Pump is provided to ensure that cooling water pressure inside the Containment fan cooler units is higher than Containment pressure during a LOCA. This prevents leakage of Containment radioactivity into the ESW system. An orifice downstream of the fan cooler units provides increased system resistance during booster pump operation. The booster pump is placed in service by an SI or LOSP sequencer actuation. Start of the booster pump causes the orifice to be placed into service by closing the orifice bypass valve. Flow bypasses the booster pump and orifice during normal plant operation.

If the ESW Booster pump trips then the function of providing increased system resistance is not occurring therefore isolating the Containment Fan Cooler will prevent an unmonitored release from occurring.

A. Correct.

B. Incorrect. Plausible because containment temperature will be signifcant at 28 psig and is could be mistaken that lower ESW flowexists (lower ESW pressure does actually exist) with the ESW Booster pump tripped.

C. Incorrect. Plausible because damage to containment fan cooler ESW piping has occured when ESW is lost and restarted on a depressurized header. This could be construed to apply to the ESW Booster pump.

D. Incorrect. Plausible because infusion of the containment atmosphere into the ESW system is possible (as with the correct answer).

Notes

000062 Loss of Nuclear Svc Water / 4

062AK3.03 Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: Guidance actions contained in EOP for Loss of nuclear service water
(CFR 41.4, 41.8 / 45.7)

Importance Rating:	4.0 4.2
Technical Reference:	FRP-J.1 Step Deviation Document (SSD-FRP-J-1) Page 3 Rev 15, Service Water Student Text
References to be provided:	None
Learning Objective:	SWS Objective 2.d
Question Origin:	OIT Dev Bank EOP-3.13-17
Comments:	None
Tier/Group:	T1G1

14. 2011 NRC RO 014/NEW/C/A/////065 AG2.2.44/

Given the following plant conditions:
- The plant is operating at 100% power
- Instrument Air is aligned in SEQUENCE 2 with all Air Compressors available

The following alarms are received sequentially within 1 minute of each other:
- ALB-002-8-5, COMPUTER ALARM AIR SYSTEMS
- ALB-002-8-4, SERVICE AIR LOW PRESS
- ALB-002-8-1, INSTRUMENT AIR LOW PRESS

Assuming Instrument Air Header pressure is at the setpoint of the last alarm received, which of the following (1) describes the expected operation of the Instrument Air Compressors and (2) the Instrument Air header pressure that would shut 1SA-506, Service Air Header Isolation Valve?

A. (1) 1A and 1B running ONLY;

(2) 90 psig

B. (1) 1A and 1B running ONLY;

(2) 75 psig

C✓ (1) 1A, 1B, and 1C running;

(2) 90 psig

D. (1) 1A, 1B, and 1C running;

(2) 75 psig

Feedback

Plausibility and Answer Analysis

At 75 psig IA header pressure (alarm setpoint) 1A and 1B air compressors will be started by CAS and 1C air compressor will start on its internal pressure switch at 101 psig. 1SA-506 automatically shuts at 90 psig IA header pressure.

A. Incorrect. Plausible since 1C air compressor is isolated from CAS in sequence 2 and only 1A and 1B air compressors are controlled by CAS.

B. Incorrect. Plausible since 1C air compressor is isolated from CAS in sequence 2 and only 1A and 1B air compressors are controlled by CAS. 75 psig is the alarm setpoint.

C. Correct.

D. Incorrect. Plausible since 75 psig is the alarm setpoint.

Notes

000065 Loss of Instrument Air / 8

065AG2.2.44
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.
(CFR: 41.5 / 43.5 / 45.12)

Importance Rating:	4.2 4.4
Technical Reference:	OP-151.01 Precaution and Limitation 6, Page 7, Rev. 62, APP-ALB-002 Page 38, Rev. 43
References to be provided:	None
Learning Objective:	Instrument and Service Air Objective 2.f
Question Origin:	NEW
Comments:	None
Tier/Group:	T1G1

15. 2011 NRC RO 015/NEW/C/A/////077 AG2.2.37/

Given the following plant conditions:

- The plant is operating at 50% power with 'B' Safety Train equipment in service
- The crew is implementing AOP-028, Grid Instability, due to degraded grid voltage
- 'A' Emergency bus has been energized from the 'A' EDG IAW AOP-028

Which ONE of the following is correct related to equipment operability under the current conditions?

A. RM-3502A-SA, Cnmt RCS Leak Detection Monitor, is inoperable due to a CVIS.

B.✓ Both Containment Vacuum Breakers are inoperable due to a CVIS.

C. Only the A-SA Train Containment Vacuum Breaker is inoperable due to a CVIS.

D. Only the A-SA Train Containment Vacuum Breaker is inoperable due to a failed closed containment isolation valve on loss of power.

Feedback

Plausibility and Answer Analysis

A. Incorrect. Plausible because RM-3502A-SA is made Inoperable during the transfer of power from off site to the 'A' EDG. However, the inoperability is caused by a temporary loss of power to the monitor. The monitor is not impacted by the CVIS.

B. Correct.

C. Incorrect. Plausible because there is only an 'A' train CVIS signal generated during the transfer of power, however it impacts the operability of both 'A' and 'B' train Containment vacuum breakers.

D. Incorrect. Plausible because the 'A-SA' Containment vacuum breaker is Inoperable from the CVIS that must be reset. The inoperability is not caused by a loss a loss of power failing closed a CIV.

Notes

000077 Generator Voltage and Electric Grid Disturbances / 6

077AG2.2.37 Ability to determine operability and/or availability of safety related equipment.
(CFR: 41.7 / 43.5 / 45.12)

Importance Rating:	3.6 4.6
Technical Reference:	AOP-028 Page 11 Rev. 28
References to be provided:	None
Learning Objective:	AOP-LP-3.28 Objective 3
Question Origin:	NEW
Comments:	None
Tier/Group:	T1G1

16. 2011 NRC RO 016/NEW/C/A/////WE04 EA2.1/

Given the following plant conditions:

- A Reactor Trip and Safety Injection has occured
- PATH-1 is being implemented and SI has been reset
- RCS Pressure is 1500 psig and stable
- PZR level is off scale low
- Subcooling is 3°F
- Containment pressure 0.2 psig
- Rad monitor, RM-1RR-3597, RHR Pump 1B is in high alarm and trending up
- Window 6-3, RAB Equip C/D Sump Alert Lvl, is lit on MLB-4A-SA and MLB-4B-SB
- SG levels are: A - 23%, B - 24%, C - 15%
- The operator has reduced total AFW flow to 215 KPPH

Based on this information, which ONE of the following procedures will be implemented when exiting PATH-1?

A. FRP-H.1, Response to Loss of Secondary Heat Sink

B.✓ EPP-013, LOCA Outside Containment

C. EPP-008, SI Termination

D. EPP-009, Post LOCA Cooldown and Depressurization

Feedback

Plausibility and Answer Analysis

The transition to EPP-013 is correct. The Radiation Monitor in alarm and sump level alert lights indicate that the leak is in the B RHR Pump Room. LOCA outside containment. Transition to EPP-013 would occur at step 38 of PATH-1.

A. Incorrect. *Plausible because S/G levels are all less than 25%, which meet FRP-H.1 entry conditions (Containment conditions normal), if total feed flow is less than 210KPPH.*

B. Correct.

C. Incorrect. *Plausible if the applicant incorrectly determines that SI termination criteria have been met. PZR level does not meet the requirement for SI termination.*

D. Incorrect. *Plausible because this is the procedure that would be implemented for the question conditions if Auxiliary building radiation levels were normal.*

Notes

W/E04 LOCA Outside Containment / 3

WE04EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Importance Rating:	3.4 4.3
Technical Reference:	PATH-1 step 38 Rev. 30
References to be provided:	None
Learning Objective:	EOP-LP-2.3/3.3 Objective 1.d
Question Origin:	NEW
Comments:	None
Tier/Group:	T1G1

17. 2011 NRC RO 017/NEW/FUNDAMENTAL////EARLY SUBMITTAL/WE11 EK3.4/
Why does EPP-012, Loss of Emergency Coolant Recirculation take precedence over FRP-J.1, Response to High Containment Pressure for operation of the Containment Spray Pumps?

The reason is based on ____________________.

A. maintaining Containment heat removal.

B.✓ conserving RWST inventory.

C. limiting Containment pressure.

D. maintaining Containment iodine removal.

Feedback

Plausibility and Answer Analysis

A step in FRP-J.1 directs operation of containment spray pumps to be IAW EPP-012 if it is in affect. EPP-012 may direct securing all containment spray pumps to conserve RWST inventory.

A. Incorrect. Plausible since EPP-012 uses the number of of cnmt fan coolers operating (heat removal capability) in evaluating the number of containment spray pumps to run.

B. Correct.

C. Incorrect. Plausible since EPP-012 uses containment pressure in evaluating the number of containment spray pumps to run.

D. Incorrect. Plausible since containment iodine removal is a function of the containment spray pumps and is a factor in determination of securing containment spray pumps in PATH-1.

Notes

W/E11 Loss of Emergency Coolant Recirc. / 4

WE11EK3.4 Knowledge of the reasons for the following responses as they apply to the (Loss of Emergency Coolant Recirculation) RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

Importance Rating: 3.6 3.8

Technical Reference: FRP-J.1 Background

References to be provided: None

Learning Objective: EOP-LP-2.3/3.3 Objective 1.c

Question Origin: New

Comments: None

Tier/Group: T1G1

18. 2011 NRC RO 018/MODIFIED/C/A////WE05 EK2.1/
Given the following plant conditions:
- A loss of Feedwater has resulted in a Reactor Trip
- The crew is performing actions of FRP-H.1, Response to Loss of Secondary Heat Sink.

Which ONE of the following describes the operation of the PZR PORVs during this event?

A. Will be operated manually to depressurize the RCS so that Low Steam Pressure and Low PZR Pressure SI signals can be blocked prior to resetting FWIS to allow establishing Main Feedwater flow.

B.✓ Will be operated manually to depressurize the RCS so that Low Steam Pressure and Low PZR Pressure SI signals can be blocked prior to performing action to establish Condensate flow.

C. Manually open all PZR PORVs to depressurize the RCS during Bleed and Feed.

D. All PZR PORVs remain in automatic during Bleed and Feed.

Feedback

Plausibility and Answer Analysis

FRP-H.1 attempts to establish Feedwater flow to the SG' with AFW first, Main FW flow next, then depressurize the RCS to between 1900 psig and 1950 psig to block the Low PZR pressure and Low Steam Presure signals. If letdown is not in service the RNO action is to use one PZR PORV's reduce pressure. The next procedure action is to then depressurize one SG to < 500 psig and establish Condensate flow.

A. Incorrect *Plausible since FRP-H.1 attachment 2,* ESTABLISHING MAIN FW FLOW TO SGs, *contains directions to Reset SI, then FWIS to allow MFW flow restoration.*

B. Correct

C. Incorrect *Plausible because establishing RCS bleed path has the operator open PZR PORVs but only two are opened per Caution prior to step 21 to minimize loss of RCS inventory.*

D. Incorrect *Plausible because the valves are allowed to remain in auto throughout the procedure for RCS overpressure concerns but are used in two cases, the first to depressurize the RCS inorder to block the Low PZR pressure and Low steam pressure SI signals and the second to establish a RCS bleed path where TWO PZR PORVs are opened (not three).*

Notes

BW/E04; W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4

WE05EK2.1 Knowledge of the interrelations between the (Loss of Secondary Heat Sink) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Importance Rating:	3.7 3.9
Technical Reference:	FRP-H.1 Step 9 page 18, 52, 53 Rev. 26
References to be provided:	None
Learning Objective:	EOP-LP-3.11 Objective 1.a
Question Origin:	OIT Dev Bank E05 EK1.1 3
Comments:	None
Tier/Group:	T1G1

19. 2011 NRC RO 019/MODIFIED/C/A/////001 AK2.01/

Given the following plant conditions:

Time = 1000

- Reactor Power is 75% and stable
- Tavg - Tref deviation is 0°F and stable
- PZR level is 51% and stable
- Control Bank D step counters are at 165 steps

Time =1002

- Reactor Power is approximately 76% and rising (no load change is in progress)
- Tavg - Tref deviation is approximately +2°F and rising
- PZR level 52% and rising
- PZR spray valves have throttled open
- Control Bank D step counters are at 174 steps and stepping out at 8 steps per minute

Which ONE of the following describes (1) the event in progress AND (2) the FIRST action that must be performed IAW AOP-001, Malfunction of Rod Control and Indication System?

A. (1) selected first stage pressure channel high failure

(2) Trip the Reactor and go to EOP PATH-1, Reactor Trip or Safety Injection

B. (1) selected first stage pressure channel high failure

(2) Place the Rod Bank Selector Switch to MAN.

C. (1) Continuous Spurious Control Bank Withdrawal

(2) Trip the Reactor and go to EOP PATH-1, Reactor Trip or Safety Injection.

D.✓ (1) Continuous Spurious Control Bank Withdrawal

(2) Place the Rod Bank Selector Switch to MAN.

Feedback

Plausibility and Answer Analysis

During Control rod withdrawl Tavg will increase due to the addition of positive reactivity. The stem of the question states that intial control rod position is 165 steps and final position is 174 steps. Therefore a Control Rod withdrawl is taking place. With a +2°F Tavg - Tref deviation the Control rods in auto should be stepping in. Immediate actions for AOP-001 are to verify that < 2 rods have dropped and then place rods in Manual and verify that rod motion is stopped.

A. Incorrect. Plausible because this failure will cause rods to move out both for temperature mismatch and for power rate mismatch. Incorrect because the indicated Tave-Tref deviation would be negative. To trip the reactor at this point would be incorrect, AOP-001 actions are required.

B Incorrect. Plausible because this failure will cause rods to move out both for temperature mismatch and for power rate mismatch. Incorrect because the indicated Tave-Tref deviation would be negative. Plausible since the second part is correct IAW AOP-001 for a continuous rod withdrawal, due to a failure if an input to Rod Control.

C. Incorrect. Plausible since this is the correct accident, however, the action stated is the RNO if rods do not cease moving once they have been placed in manual IAW AOP-001.

D. Correct.

Notes

000001 Continuous Rod Withdrawal / 1

001AK2.01
Knowledge of the interrelations between the Continuous Rod Withdrawal and the following:Rod bank step counters
(CFR 41.7 / 45.7)

Importance Rating:	2.9 3.2
Technical Reference:	AOP-001 immediate actions step 2, Rev 35
References to be provided:	None
Learning Objective:	AOP-LP-3.1 Objective 4
Question Origin:	Modified
Comments:	None
Tier/Group:	T1G2

20. 2011 NRC RO 020/NEW/C/A/////028 AK2.03/

Given the following plant conditions:

- The plant is at 60% power with T-avg at T-ref
- The PZR level channel selector switch (LS-459Z) is in the 460/461 position

Which ONE of the following describes the impact of PZR level Transmitter LT-460 failing to 55% with these conditions?

(Assume PZR pressure remains constant)

	FK-122.1 (1CS-231) Charging Flow Controller Output	PZR backup heaters
A.	Remains the same	energize
B✓	Remains the same	remain off
C.	Decreases	energize
D.	Decreases	remain off

Feedback

Plausibility and Answer Analysis

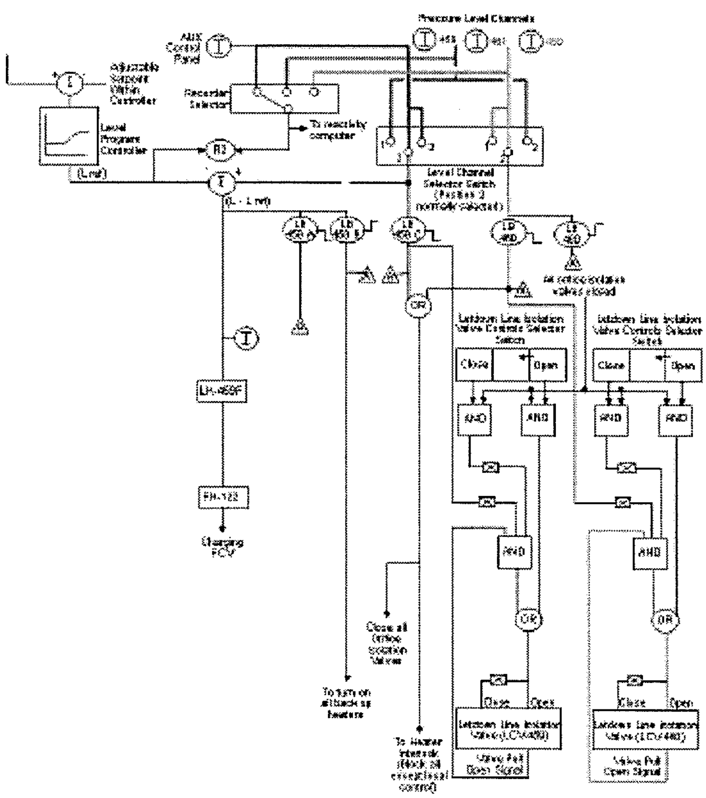
In the 460/461 position, LT-461 provides the input to PZR Level Controller LK-459F (in the place of LT-459) which in turn provides the input to FK-122.1. LT-461 also provides the input to the high level deviation function (in the place of LT-459). LT-460 only provides inputs to the LT-460 functions such as the letdown isolation and heater cutoff at 17% level. Failure of LT-460 to 55% does not impact level control or deviation. Program level is 46% at 60% power.

A. *Incorrect.* *Plausible because the output remains the same and PZR heaters would turn on on a 5% level deviation with LT-461.*

B. *Correct.*

C. *Incorrect.* *Plausible because the output would decrease and heaters turn on if LT-460 was the controlling channel.*

D. *Incorrect.* *Plausible because the output would decrease if LT-460 was the controlling channel and high level deviation to turn PZR heaters on could be unknown or miscalculated.*



Notes

000028 Pressurizer Level Malfunction / 2

028AK2.03 Knowledge of the interrelations between the Pressurizer Level Control Malfunctions and the following: Controllers and positioners
(CFR 41.7 / 45.7)

Importance Rating:	2.9 3.2
Technical Reference:	APP-ALB-009-2-1, Rev. 13
References to be provided:	None
Learning Objective:	PZRLC Objective 3.a
Question Origin:	NEW
Comments:	None
Tier/Group:	T1G2

21. 2011 NRC RO 021/MODIFIED/FUNDAMENTAL/////037 AA1.02/

Given the following plant conditions:
- A SG tube leak is in progress on the 'B' SG
- HP estimated the tube leak to be 155 gallons per day
- The crew is implementing AOP-016, Excessive Primary Plant Leakage
- Turbine Building Vent Stack and CVPETS Rad Monitor are in high alarm

Which ONE of the following describes the operation of CVPETS?

A. A CVPETS Exhaust Fan started automatically on high radiation and all dampers repositioned automatically on the fan start.

B. A CVPETS Exhaust Fan started automatically on high radiation and the dampers are normally aligned for operation.

C✓ A CVPETS Exhaust Fan is started manually on high radiation and all dampers repositioned automatically on fan start.

D. A CVPETS Exhaust Fan is started manually on high radiation and the dampers are normally aligned for operation.

Feedback

Plausibility and Answer Analysis

During a Primary to Secondary leak AOP-016, "Excessive Primary Plant Leakage" is used to provide guidance. When both the Turbine Building Vent Stack radiation monitor is in High Alarm AND SG tube leakage is < Tech Spec limits (found in TS 3.4.6.2.c - 150 gallons per day through any one SG) then in Attachment 1 step 6 RNO the operator is directed to start CVPETS IAW OP-133, Main Condenser Air Removal System. The control switch for CVPETS is located on AEP-2. There are no auto start exhaust fan (E-79A or E-79B) functions associated with CVPETS, they must be manually operated. 1AE-23, CVPETS bypass AE-B3 will auto shut and the associated CVPETS dampers do automatically position following an exhaust fan start.

A. Incorrect. — *Plausible because while this fan does not start on high radiation, other fans in the plant do. Examples are the Fuel Handling Building Emergency Exhaust and MCR Recirculation Fans. The dampers do reposition on fan start.*

B. Incorrect — *Plausible because while this fan does not start on high radiation, other fans in the plant do. Examples are the Fuel Handling Building Emergency Exhaust and MCR Recirculation Fans. Most plant systems with auto starts are aligned for operation requiring only the start.*

C. Correct

D. Incorrect. — *The First part is correct. Second part is plausible because most plant systems with auto starts are aligned for operation requiring only the start.*

Notes

000037 Steam Generator Tube Leak / 3

037AA1.02 Ability to operate and / or monitor the following as they apply to the Steam Generator Tube Leak: Condensate exhaust system
(CFR 41.7 / 45.5 / 45.6)

Importance Rating:	3.1 2.9
Technical Reference:	AOP-016, Attachment 1 Step 6 page 18 Rev. 43, OP-133, Section 8.1 step 2 Rev. 39
References to be provided:	None
Learning Objective:	MCES-TP-3.0 Obj 5
Question Origin:	OIT Dev Bank MCES-R5 1
Comments:	None
Tier/Group:	T1G2

22. 2011 NRC RO 022/BANK/C/A////051 AK3.01/

Given the following plant conditions:

- The plant is operating at 100% power
- A complete loss of Zone 1 and Zone 2 vacuum occurs

Which ONE of the following will (1) automatically operate to stabilize RCS temperature instead of Condenser Steam Dumps AND (2) why does C-9 prevent Condenser Steam Dump operation?

A. (1) Atmospheric Steam Dumps

(2) Prevent over-heating of the condenser boot seals

B. (1) Atmospheric Steam Dumps

(2) Protect the Condenser from an overpressure condition

C. (1) Steam Generator PORVs

(2) Prevent over-heating of the condenser boot seals

D✓ (1) Steam Generator PORVs

(2) Protect the Condenser from an overpressure condition

Feedback

Plausibility and Answer Analysis

The condenser steam dumps must have the Condenser availabe in order to be used. A loss of Condenser vaccuum would prevent the Condenser steam dumps from being used. On a Turbine Trip the atmospheric steam dumps do not actuate. Therefore the next option available for dumping steam would be the SG PORV's followed by the SG Safety valves. The reason that the Condenser steam dumps cannot be used is Control interlock C-9 is not made up (at least 1 Circ Water pump breaker is closed and Condenser vaccuum must initially be established better than 24.5 " Hg vacuum in both condenser zones and remain better than 22.5" Hg vaccuum in both zones). If steam were dumped into the Condenser without C-9 met then a Condenser overpressure condition could occur causing severe damage. SG PORVs would normally be set to open at 1105 psig (controller setpoint of 85%). SG safety valve setpoints are 1170, 1185, 1215 and 1230 psig.

A. Incorrect. *Plausible since the condenser dumps are not able to respond to control temperature on a Turbine trip due to the loss of C-9, but the atmospheric steam dumps do not get an arming signal on a Turbine trip. Second part is plausible, as main condenser temperature will go up and affect boot seal temperature, but this is not the reason for C-9.*

B. Incorrect. *Plausible since the condenser dumps are not able to respond to control temperature on a Turbine trip due to the loss of C-9, but the atmospheric steam dumps do not get an arming signal on a Turbine trip.*

C. Incorrect. *Plausible since the SG PORVs will actuate on a Turbine trip due to a loss of vacuum, but it is to prevent an overpressure condition of the condenser. Main condenser temperature will go up and affect boot seal temperature, but this is not the reason for C-9.*

D. Correct.

Notes

000051 Loss of Condenser Vacuum / 4

051AK3.01 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
(CFR 41.5,41.10 / 45.6 / 45.13)

Importance Rating:	2.8* 3.1*
Technical Reference:	MSSS Lesson Plan
References to be provided:	None
Learning Objective:	MSSS Objective 6.c
Question Origin:	OIT Dev Bank SDCS-R6
Comments:	None
Tier/Group:	T1G2

23. 2011 NRC RO 023/BANK/FUNDAMENTAL////EARLY SUBMITTAL/059 AK3.01/
Given the following plant conditions:
- A liquid release was planned from the Treated Laundry and Hot Shower (TL&HS) Tank 'A'
- A resulting Human Performance error inadvertantly lined up the non-sampled TL&HS Tank 'B' for release
- During the release REM-1WL-3540, TL&HS Tank Pump Discharge went into HIGH Alarm

Which ONE of the following describes (1) how the accidental release is terminated AND (2) the reason for terminating the release?

A. (1) the running TL&HS Tank Pump will automatically trip

(2) to prevent contamination of the cooling tower basin

B. (1) 3LHS-296, TREATED L&HS TKS DISCH ISOL VLV, will automatically close

(2) to prevent contamination of the cooling tower basin

C. (1) the running TL&HS Tank Pump will automatically trip

(2) to ensure the release is less than the regulatory limits

D✓ (1) 3LHS-296, TREATED L&HS TKS DISCH ISOL VLV, will automatically close

(2) to ensure the release is less than the regulatory limits

Feedback

Plausibility and Answer Analysis

AOP-005 has the operator verify that valve 3LHS-296, Treated LHS Tk Disch Isol Valve shuts upon high alarm for REM-1WL-3540. The purpose of the auto isolation is to ensure that liquid effulent releases are maintained less than the limits specified in 10CFR20.

A. Incorrect. Plausible since this would terminate the release, however RM-3540 does not trip the TL&HS tank pump it only closes 3LHS-296. Different Rad Monitors such as RM-3528 Tank Areas drains will trip their associated pump.

Also, the basis for securing the release is to prevent from exceeding release limits not contaminating the Cooling tower basin.

B. Incorrect. Plausible since a high alarm on RM-3540 will trip 3LHS-296 shut. However the basis for securing the release is to prevent from exceeding regulatory limits not contaminating the Cooling tower basin.

C. Incorrect. Plausible since this would terminate the release, however RM-3540 does not trip the TL&HS tank pump it only closes 3LHS-296. Different Rad Monitors such as RM-3528 Tank Areas drains will trip their associated pump.

D. Correct.

Notes

000059 Accidental Liquid RadWaste Rel. / 9

059AK3.01 Knowledge of the reasons for the following responses as they apply to the Accidental Liquid Radwaste Release: Termination of a release of radioactive liquid (CFR 41.5,41.10 / 45.6 / 45.13)

Importance Rating:	3.5 3.9
Technical Reference:	AOP-005 Attachment 9 step 4 Page 26 Rev 27, AOP-008-BD discussion item Page 3 Rev. 2
References to be provided:	None
Learning Objective:	AOP-LP-3.8 Objective 3
Question Origin:	Bank OIT Dev
Comments:	None
Tier/Group:	T1G2

24. 2011 NRC RO 024/PREVIOUS/FUNDAMENTAL/////068 AA2.03/

Given the following plant conditions:

- An electrical fire in the MCR has resulted in the evacuation of the MCR to the ACP
- You have been directed to monitor in core thermocouple temperatures from outside the MCR per AOP-004, Remote Shutdown

Which ONE of the following (1) identifies the location of the Inadequate Core Cooling Monitor local microprocessor panel AND (2) the method of obtaining readings at the panel?

A. (1) PIC Room C-17 RAB 286'

(2) Using CRT monitor and keyboard.

B.✓ (1) PIC Room C-17 RAB 286'

(2) Using thumbwheels set to specific points identified by a legend.

C. (1) Main Termination Cabinet RAB 305'

(2) Using CRT monitor and keyboard.

D. (1) Main Termination Cabinet RAB 305'

(2) Using thumbwheels set to specific points identified by a legend.

Feedback

Plausibility and Answer Analysis

'A' Train RVLIS cabinet is located on at PIC Room C-17 RAB 286' and the reading would be obtained by moving a thumbwheel to different points with the readouts identified by a placard on the inside of the cabinet door.

A. Incorrect. *Plausible because the location is correct and the method to obtain a reading is correct if the reading was obtained in the Main Control Room not locally.*

B. Correct.

C. Inncorrect. *Location is incorrect but plausible because this is the location of the 'B' Train cabinet. The method used to obtain a reading is incorrect but plausible because this is the method used to obtain a reading on the panel in the Main Control Room.*

D is incorrect. *Location is incorrect but plausible because this is the location of the 'B' Train cabinet. The method to obtain the display readings are correct.*

Lower cognitive level item requires applicant to know location and equipment available at the location

Notes

000068 (BW/A06) Control Room Evac. / 8

068AA2.03 Ability to determine and interpret the following as they apply to the Control Room Evacuation:T-hot, T-cold, and in-core temperatures
(CFR: 43.5 / 45.13)

Importance Rating:	4.0 4.2
Technical Reference:	AOP-004, rev. 55, section 3.1 step 37.c, pp 43, OWP-PAM for locations of cabinets
References to be provided:	None
Learning Objective:	Inadequate Core Cooling Monitor Objective 5
Question Origin:	Bank
Comments:	**Previous** 2009B NRC RO question
Tier/Group:	T1G2

25. 2011 NRC RO 025/NEW/C/A/////074 EA2.07/

Given the following plant conditions:

- A cold leg break LOCA has occurred and the crew is implementing PATH-1
- ALL RCPs are stopped
- Containment pressure is 8 psig
- ERFIS is NOT available

Time	Core Exit TCs	RCS Pressure	RVLIS Full Range
0000	450°F	800 psig	55%
0030	550°F	900 psig	45%
0100	700°F	700 psig	35%
0130	800°F	600 psig	25%
0200	1220°F	500 psig	20%

Based on the values of the Core Exit TCs, RCS pressure, and RVLIS Full Range level above at given times; at what time does the transition to:

1.) *degraded* core cooling first occur.
2.) *inadequate* core cooling first occur.

A. (1) 0030

(2) 0130

B. (1) 0030

(2) 0200

C.✓ (1) 0100

(2) 0130

D. (1) 0100

(2) 0200

Feedback

Plausibility and Answer Analysis

Degraded core cooling exists when either core exit TCs are greater than 730°F or RVLIS Full Range level is less than 39%. Inadequate core cooling exists when either both core exit TCs are greater than 730°F and RVLIS Full Range level is less than 39% or when core exit TCs are greater than 1200°F. Both core exit TCs are greater than 730°F and RVLIS Full Range level is less than 39% was reached prior to 1200°F.

A. Incorrect. The first part is plausible since saturated core cooling exist at 0030 (subcooling is less than 50°F) which could be confused with degraded core cooling. The second part is correct.

B. Incorrect. The first part is plausible since saturated core cooling exist at 0030 (subcooling is less than 50°F) which could be confused with degraded core cooling. The seond part is plausible since inadequate core cooling does exist based on core exit TCs alone but it was reached earlier based on core exit TCs and RVLIS level.

C. Correct.

D. Incorrect. The first part is correct. The seond part is plausible since inadequate core cooling does exist based on core exit TCs alone but it was reached earlier based on core exit TCs and RVLIS level.

Notes

000074 (W/E06&E07) Inad. Core Cooling / 4

074EA2.07 Ability to determine or interpret the following as they apply to a Inadequate Core Cooling: The difference between a LOCA and inadequate core cooling, from trends and indicators
(CFR 43.5 / 45.13)

Importance Rating:	4.1 4.7
Technical Reference:	Critical Safety Funtion -2 - Core Cooling
References to be provided:	None
Learning Objective:	EOP-LP-3.10 Objective 4.a
Question Origin:	NEW
Comments:	None
Tier/Group:	T1G2

26. 2011 NRC RO 026/BANK/C/A/////WE03 EG2.1.32/

Given the following plant conditions:

- At 0530, RCS temperature was being maintained at 557°F with the plant in Mode 3 when a Small Break LOCA occurred
- At 0545, RCS temperature is 550°F
- The crew is ready to commence a cooldown to cold shutdown IAW EPP-009, Post LOCA Cooldown and Depressurization

Which ONE of the following identifies (1) the lowest allowable temperature of the RCS at 0630 if the crew begins the MAXIMUM permissible cooldown rate AND (2) the basis for this temperature limit?

A.✔ (1) 457°F

(2) to ensure that Tech Spec cooldown limits are NOT exceeded

B. (1) 457°F

(2) to ensure that a transition is NOT required to be made to FRP-P.1, Response to Imminent Pressurized Thermal Shock

C. (1) 450°F

(2) to ensure that Tech Spec cooldown limits are NOT exceeded

D. (1) 450°F

(2) to ensure that a transition is NOT required to be made to FRP-P.1, Response to Imminent Pressurized Thermal Shock

Feedback

Plausibility and Answer Analysis

In determining the RCS cooldown rate, the cold leg temperature change over the last 60 minutes must always be considered to ensure TS limits are not exceeded so the lowest temperature at 0630 with a 100°F CD will be 457°F.

A. Correct

B. Incorrect — *Plausible since the lowest allowed temperature is 457°F, but it is to ensure TS limits are not exceeded.*

C. Incorrect — *Plausible since if a cooldown rate of 100°F per hour over 45 minutes is performed 450°F will be the resulting temperature, but the cold leg temperature change over the last 60 minutes must always be considered to ensure TS limits are not exceeded.*

D. Incorrect — *Plausible since if a cooldown rate of 100°F per hour over 45 minutes is performed 450°F will be the resulting temperature, but the cold leg temperature change over the last 60 minutes must always be considered to ensure TS limits are not exceeded.*

Notes

BW/E08; W/E03 LOCA Cooldown - Depress. / 4

WE03EG2.1.32 Ability to explain and apply system limits and precautions.
(CFR: 41.10 / 43.2 / 45.12)

Importance Rating:	3.8	4.0
Technical Reference:	EOP Users Guide, Rev. 30, Page 46; EPP-009 Step 10 Page 12 Rev. 18	
References to be provided:	None	
Learning Objective:	EOP-LP-3.5 Objective 4	
Question Origin:	OIT Dev Bank EOP-3.5-R5	
Comments:	None	
Tier/Group:	T1G2	

27. 2011 NRC RO 027/BANK/C/A/////WE09 EK1.2/

Given the following plant conditions:
- A Reactor trip occurred due to a Loss of Offsite Power
- The plant is being cooled down on RHR per EPP-006, Natural Circulation Cooldown with Steam Void in Vessel With RVLIS
- RCS cold leg temperatures are 190°F
- Steam generator pressures are <50 psig
- RVLIS upper range indicates greater than 100%
- Two CRDM fans have been running during the entire cooldown

IAW EPP-006, steam should be dumped from all SGs to ensure . . .

A. RCS temperatures do NOT increase during the required 29 hour vessel soak period.

B. boron concentration is equalized throughout the RCS prior to taking a sample to verify cold shutdown boron conditions.

C✓ all inactive portions of the RCS are below 200°F prior to complete RCS depressurization.

D. secondary side water temperatures are less than 50°F above RCS cold leg temperatures prior to any subsequent RCP restart if off-site power is restored.

Feedback

Plausibility and Answer Analysis

SG pressure above 0 psig indicates that the SGs are above 200°F (<50 psig is below last MCB meter increment). Depressurizing the RCS under this condition will result in additional void formation in the SG U-tubes.

A. Incorrect. *Plausible since a soak period is addressed, but only if continued operation of CRDM fans had not been maintained.*

B. Incorrect. *Plausible since this action would have been performed in this procedure, but must be completed prior to depressurizing the RCS below 1900 psig.*

C. Correct.

D. Incorrect. *Plausible since RCP operation throughout NC Cooldown is desirable and the TS limit on the temperature differential between the SGs and RCS is applicable. Incorrect since temperature differential is swapped.*

Notes

BW/E09; CE/A13; W/E09&E10 Natural Circ. / 4

WE09EK1.2 Knowledge of the operational implications of the following concepts as they apply to the (Natural Circulation Operations) Normal, abnormal and emergency operating procedures associated with (Natural Circulation Operations

Importance Rating:	3.3 3.7
Technical Reference:	EPP-006 Step 12 Page 18, Rev. 13 and EPP-006 Step 13, Page 20, Rev. 13
References to be provided:	None
Learning Objective:	EOP-LP-3.8 Objective 2.e
Question Origin:	Bank
Comments:	None
Tier/Group:	T1G2

28. 2011 NRC RO 028/NEW/C/A/////003 G2.2.42/

Given the following plant conditions:

At 0955

- A Unit startup is in progress IAW GP-004, Reactor Startup (Mode 3 to Mode 2)
- Reactor trip breakers are CLOSED
- All Shutdown banks are fully inserted
- Annunciator ALB-010-2-3, RCP-A Trouble, alarmed
- AOP-018, Reactor Coolant Pumps Abnormal Conditions was entered

At 1005 - the crew secured 'A' RCP when its motor bearing temperatures approached the pump trip limit

At 1020 - Start Up XFMR A to Aux Bus A breaker 107 tripped open

Which ONE of the following completes the statement below?

At __(1)__ Tech Spec 3.4.1.2 LCO is NOT met and the Reactor Trip breakers must be opened __(2)__ per the Tech Spec required action.

A. (1) 1005

(2) immediately

B. (1) 1005

(2) within 1 hour

C. (1) 1020

(2) immediately

D✓ (1) 1020

(2) within 1 hour

Feedback

Plausibility and Answer Analysis

During the performace of GP-004, a mode change from Mode 3 to Mode 2 takes place AFTER the shutdown banks are withdrawn. Therefore the plant is still in Mode 3 and TS LCO's would apply for Mode 3. At 1005 one RCP is inoperable and TS 3.4.1.2 is still met (2 RC loops are still OPERABLE) with Reactor Trip breakers closed.

At 1020 Aux Bus A is deenergized. Normal plant electrical lineup has Aux Bus A is crosstied with Aux Bus C. Aux Bus C only has ONE load which is the 'C' RCP. Therefore at 1020 with the plant in Mode 3 and both the 'A' and 'C' RCP inoperable TS 3.4.1.2 action b would apply. - With only one RC loop in operation and the Reactor Trip System breakers in the closed position, within 1 hour open the Reactor Trip System breakers.

A. Incorrect. *Plausible because if the plant was in Modes 1 or 2 at 1005 the loss of ONE RCP would be entry condtions into TS 3.4.1.1 but would not require the plant to have the Reactor trip breakers open (be in HSB) for 6 hours.*

B Incorrect. *Plausible because if the plant was in Modes 1 or 2 at 1005 the loss of ONE RCP would be entry condtions into TS 3.4.1.1 but would not require the plant to have the Reactor trip breakers open (be in HSB) for 6 hours.*

C. Incorrect. *Plausible because 1020 is the correct time but the action is incorrect. TS 3.4.1.2.c (open Reactor Trip Breakers <u>immediately</u>) would be correct if there were no RCP were in operation but at this time the 'B' RCP is still in operation.*

D. Correct.

Notes

003 Reactor Coolant Pump

003G2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications.
(CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)

Importance Rating:	3.9 4.6
Technical Reference:	Tech Specs 3.4.1.2 Action b
References to be provided:	None
Learning Objective:	TS-LP-2.0/3.0/5.0/8.0 Objective 4.e
Question Origin:	NEW
Comments:	None
Tier/Group:	T2G1

29. 2011 NRC RO 029/BANK/FUNDAMENTAL/////003 K6.14/

Given the following plant conditions:

- The plant is in Mode 4
- Preparations are underway to start the 'A' RCP

Which ONE of the following conditions does NOT meet the RCP starting requirements IAW OP-100, Reactor Coolant System?

A.✓ VCT pressure is 11 psig

B. RCS pressure is 330 psig

C. RCP #1 seal leakoff flow is 0.85 gpm

D. RCP #1 seal Delta P is 210 psid

Feedback

Plausibility and Answer Analysis

OP-100, Reactor Coolant System contains 36 Precautions and Limitations. P&L #5 applies when starting a RCP.

Volume Control Tank pressure should be maintained greater than 15 psig to insure sufficient cooling water is supplied to # 2 RCP Seal.

A. Correct.

B. Incorrect. *Plausible because it represents a presure very close to the minimum requirement for NPSH (325 psig)*

C. Incorrect. *Plausible because seal leakoff is low, but not low enough (0.2 gpm) to cause damage to seals upon start*

D. Incorrect. *Plausible because #1 seal DP is low but within the limits required (>200 psid) to ensure normal seal operation on RCP start*

Notes

003 Reactor Coolant Pump

003K6.14 Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: Starting requirements
(CFR: 41.7 / 45/5)

Importance Rating:	2.6 2.9
Technical Reference:	OP-100, 4.0.5 and section 5.1 Rev. 35
References to be provided:	None
Learning Objective:	RCS Objective 6.a
Question Origin:	Bank OIT Dev. 003 K6.14 1
Comments:	None
Tier/Group:	T2G1

30. 2011 NRC RO 030/BANK/C/A/////004 A2.22/

Given the following plant conditions:

- The plant is operating at 100% power
- The OAC provides a crew update that PZR level is trending DOWN and VCT level is trending UP
- RCS temperature and pressure are stable

Which ONE of the following describes the event in progress AND action required?

A. Charging line leak outside Containment. Isolate the leak IAW AOP-016, Excessive Primary Plant Leakage.

B. Letdown line leak outside Containment. Isolate the leak IAW AOP-016, Excessive Primary Plant Leakage.

C. Letdown pressure control valve, 1CS-38 (PCV-145), has failed shut. Isolate letdown IAW the applicable Annunciator Panel Procedures.

D✓ Charging flow controller failed to minimum. Establish Manual control of Charging flow or isolate Letdown IAW the applicable Annunciator Panel Procedures.

Feedback

Plausibility and Answer Analysis

A. Incorrect. Plausible because if there were a charging line leak PZR level would go down, however VCT level would be stable or lowering. In addition AOP-016 would be the proper procedure address the charging line leakage.

B. Incorrect. Plausible because a normal RCS leak would cause PZR level to go down, however letdown is different. Normal charging flow is not affected so PZR level remains on program and VCT level would go down.

C Incorrect. Plausible since a closed failure of PCV-145 would cause a mismatch between charging and letdown flows. The candidate will have to determine that the system reponse is incorrect for the given conditions.

D. Correct.

Notes

004 Chemical and Volume Control

004A2.22 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: Mismatch of letdown and changing flows
(CFR: 41.5/ 43/5 / 45/3 / 45/5)

Importance Rating:	3.2 3.1
Technical Reference:	APP-ALB-009-4-1 Rev. 13
References to be provided:	None
Learning Objective:	CVCS Objective 12.e
Question Origin:	OIT Dev Bank 022 AA1.06 1
Comments:	None
Tier/Group:	T2G1

31. 2011 NRC RO 031/BANK/FUNDAMENTAL/////005 K2.03/
Given the following plant conditions:
- The plant is operating in Mode 1
- 1RH-1 and 1RH-2,RHR Hot Leg Loop 1 Isolation Valves are closed

Which ONE of the following describes (1) the power supply AND (2) breaker status for 1RH-1 and 1RH-2?

A. (1) 1RH1 is powered from 1A21-SA, 1RH2 is powered from 1B21-SB.

(2) breakers are CLOSED

B. (1) 1RH1 is powered from 1A21-SA, 1RH2, is powered from 1B21-SB.

(2) breakers are OPEN

C. (1) 1RH1 is powered from 1B21-SB, 1RH2, is powered from 1A21-SA

(2) breakers are CLOSED

D✓ (1) 1RH1 is powered from 1B21-SB, 1RH2, is powered from 1A21-SA

(2) breakers are OPEN

Feedback

Plausibility and Answer Analysis

Each valve in series is powered by a different train to ensure ability to close for leak isolation purposes. In Mode 1, 1RH-1 and 1RH-2 breakers are locked open to prevent inadvertent opening.

A. Incorrect. *Plausible since these valves are for both for train A-SA RHR. Incorrect because the valves in the same train have different power supplies. Plausible since the valve positions are available in Mode 1, indicated on the control board, from an independent power source.*

B. Incorrect. *Plausible since these valves are for both for train A-SA RHR. Incorrect because the valves in the same train have different power supplies. Second part is correct.*

C. Incorrect. *First part is correct. Second part is plausible since the valve positions are available in Mode 1, indicated on the control board, from an independent power source.*

D. Correct.

Notes

005 Residual Heat Removal

005K2.03 Knowledge of bus power supplies to the following: RCS pressure boundary motor-operated valves
(CFR: 41.7)

Importance Rating:	2.7* 2.8*
Technical Reference:	OP-111 Section 7.2 Pages 34 and 37, Rev. 52
References to be provided:	None
Learning Objective:	RHR Objective 2
Question Origin:	OIT Dev Bank 005 K2.03
Comments:	None
Tier/Group:	T2G1

32. 2011 NRC RO 032/NEW/C/A/////005 K3.07/

Given the following plant conditions:

- The plant is in Mode 6 and core reload has just commenced
- 'A Train RHR is in service
- 'B' Train RHR is under clearance for breaker overhaul and not expected to be returned for service for 2 hours
- Cavity level is 23 feet 4 inches above the Reactor Vessel Flange
- RCS temperature is 80°F

- 'A' RHR Pump trips for an unknown reason

Which ONE of the following describes the actions required associated with this event?

A. Actuate Phase A Isolation

B.✓ Immediately suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System.

C. Refueling activities are permitted for up to 1 hour while repairs are initiated to 'A' RHR Pump.

D. Immediately close or verify closed all Containment penetrations providing direct access from the containment atmosphere to the outside atmosphere

Feedback

Plausibility and Answer Analysis

Both trains of RHR are inoperable. With neither RHR loop operable TS 3.9.8.1 requires the suspension of all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. The opererators should immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. It is also required to close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere (within 4 hours).

A. Incorrect. *Plausible since AOP-020 would have the operator actuate Phase A Isolation IF RCS temperature could not be maintained at or below 200°F. But, since core reload has just commenced there would be very little heat load in the core and RCS heatup to > 200° due to a loss of RHR would take many more hours than 2 (at which time the 'B' RHR pump would be restored and RCS cooling would resume.*

B. Correct.

C. Incorrect. *Plausible since Tech Specs allows both RHR loops to be removed from operation for up to 1 hour per 2-hour period during the performance of core alterations and core loading verification in the vicinity of the reactor vessel hot legs.*

D. Incorrect. *Plausible since Tech Spec 3.9.8.1 requires all Containment penetrations providing direct access from the Containment atmosphere to the outside atmosphere but the Tech Spec is NOT immediately it is within 4 hours.*

Notes

005 Residual Heat Removal

005K3.07 Knowledge of the effect that a loss or malfunction of the RHRS will have on the following: Refueling operations
(CFR: 41.7 / 45.6)

Importance Rating:	3.2* 3.6*
Technical Reference:	TS 3.9.8.1
References to be provided:	None
Learning Objective:	TS-LP-2.0/3.0/5.0/8.0 Objective 4.e
Question Origin:	NEW
Comments:	None
Tier/Group:	T2G1

33. 2011 NRC RO 033/NEW/FUNDAMENTAL/////006 A1.07/
Which ONE of the following completes the statements below?

IAW PATH-1 Foldout A:

- The minimum RCS pressure that the CSIP alternate miniflow isolation or miniflow block valves are verified closed is __(1)__ psig
- The maximum RCS pressure that the CSIP alternate miniflow isolation or miniflow block valves are verified open is __(2)__ psig

A. (1) 1400

(2) 2200

B. (1) 1400

(2) 2000

C.✓ (1) 1800

(2) 2200

D. (1) 1800

(2) 2000

Feedback

Plausibility and Answer Analysis

IF RCS pressure decreases to < 1800 psig, THEN verify alternate miniflow isolation OR miniflow block valves - SHUT. IF RCS pressure increases to > 2200 psig, THEN verify alternate miniflow isolation AND miniflow block valves - OPEN.

A. Incorrect. Plausible since 1400 psig is used in PATH-1 but for RCP trip criteria. The second part of answer is correct.

B Incorrect. Plausible since 1400 psig is used in PATH-1 but for RCP trip criteria. The 2000 psig is also used in PATH-1 but for blocking SI and for P-11 functions such as closing a failed open PORV and auto opens the Accumulator Discharge valves.

C. Correct.

D. Incorrect. Plausible since the 1800 psig pressure is correct and the 2000 psig is also used in PATH-1 but for blocking SI and for P-11 functions such as closing a failed open PORV and auto opens the Accumulator Discharge valves.

Notes

006 Emergency Core Cooling

006A1.07 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Pressure, high and low
(CFR: 41.5 / 45.5)

Importance Rating:	3.3 3.6
Technical Reference:	PATH-1 Foldout A, Rev. 30
References to be provided:	None
Learning Objective:	CVCS Objective 2.e
Question Origin:	NEW
Comments:	None
Tier/Group:	T2G1

34. 2011 NRC RO 034/BANK/FUNDAMENTAL//EPP-010///006 A4.04/

Given the following plant conditions:

- The plant has experienced a Large Break LOCA
- RWST level has decreased to 23%
- The operating crew has implemented plant procedures to transfer to Cold Leg Recirculation IAW EPP-010, Transfer to Cold Leg Recirculation

Which ONE of the following conditions is consistent with the desired lineup of the Residual Heat Removal system for this condition?

A. 1SI-359 SA, Low Head SI Trains A & B to Hot Leg is OPEN

B. 1SI-340 SA, Low Head SI Train A to Cold Leg **AND**
1SI-341 SB, Low Head SI Train B to Cold Leg are OPEN

C.✓ 1RH-25, Suction from RHR Heat Exchanger A-SA **AND**
1RH-63, Suction from RHR Heat Exchanger B-SB are OPEN

D. 1SI-326 SA, Low Head SI Train A to Hot Leg Crossover **AND**
1SI-327 SB, Low Head SI Train B to Hot Leg Crossover are SHUT

Feedback

Plausibility and Answer Analysis

A. Incorrect. Plausible because 1SI-359 is opened during Hot leg recirculation not cold leg recirculation.

B. Incorrect. Plausible because ONE Low Head SI Cold leg valve is open during cold leg recirculation. (Either 1SI-340 or 1SI-341) the other is closed.

C. Correct

D. Incorrect. Plausible since the normal lineup for cold leg recirculation has both 1SI-326 and 1SI-327 Open.

Notes

006 Emergency Core Cooling

006A4.04 Ability to manually operate and/or monitor in the control room: RHRS
(CFR: 41.7 / 45.5 to 45.8)

Importance Rating:	3.7* 3.6
Technical Reference:	EPP-010 Step 3, Page 8 Rev. 26
References to be provided:	None
Learning Objective:	RHR Objective 4.f
Question Origin:	Bank OIT Exam Bank RHR (04F) 1
Comments:	None
Tier/Group:	T2G1

35. 2011 NRC RO 035/MODIFIED/FUNDAMENTAL////2009B NRC RO/007 A1.01/
Given the following plant conditions:
- A PZR safety valve is leaking by its seat
- PRT pressure is currently 28 psig and increasing
- PRT level is 72% and increasing

IF the present conditions continue, (1) when will the PRT high level alarm first occur AND (2) IAW OP-100, Reactor Coolant System, in order to prevent the PRT from going water solid, which pump will the operator use to transfer the PRT to the RHT?

A. (1) 78%

(2) the Reactor Coolant Drain Tank Pump

B.✓ (1) 83%

(2) the Reactor Coolant Drain Tank Pump

C. (1) 78%

(2) the Boron Recycle Evaporator Feed Pump

D. (1) 83%

(2) the Boron Recycle Evaporator Feed Pump

Feedback

Plausibility and Answer Analysis

Per ALB-009-8-1 the PRT high level setpoint is 83%. When high level is reached in the PRT ALB-009-8-1 directs the operator to drain the PRT to normal level using OP-100. IAW OP-100 the PRT is pumped using the RCDT pumps to the Recycle Hold Up Tank.

A. Incorrect. Plausible if the applicant confuses the PRT high level setpoint with another level similar to the PRT high level (SG high level is 78%). The second part is correct.

B. Correct.

C. Incorrect. Plausible because 78% is the high level setpoint for Steam Generator water level and is a common level for applicant to remember and close to the actual setpoint for PRT high level. The second part is plausible because PRT inventory is transferred to the RHT and the BR Evap Feed Pump is used for RHT transfer evolutions.

D. Incorrect. The level setpoint correct. The second part is plausible because PRT inventory is transferred to the RHT and the BR Evap Feed Pump is used for RHT transfer evolutions.

Notes

007 Pressurizer Relief/Quench Tank

007A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including: Maintaining quench tank water level within limits
(CFR: 41.5 / 45.5)

Importance Rating:	2.9 3.1
Technical Reference:	APP ALB-009-8-1 Rev. 12 OP-100 Rev. 31
References to be provided:	None
Learning Objective:	PZR Objective 5
Question Origin:	Bank
Comments:	**Previous** 2009B NRC RO questiom
Tier/Group:	T2G1

36. 2011 NRC RO 036/NEW/FUNDAMENTAL/////008 K2.02/

Which ONE of the following describes (1) the power supplies that the 'C' CCW pump can be aligned to AND (2) which design feature prevents two CCW pumps from being aligned to the same power supply?

A.✓ (1) Either Emergency Bus 1A-SA or Emergency Bus 1B-SB

(2) A key-operated interlock

B. (1) Either Emergency Bus 1A2-SA or Emergency Bus 1B2-SB

(2) A key-operated interlock

C. (1) Either Emergency Bus 1A-SA or Emergency Bus 1B-SB

(2) A manual transfer switch must be aligned to the appropriate power supply.

D. (1) Either Emergency Bus 1A2-SA or Emergency Bus 1B2-SB

(2) A manual transfer switch must be aligned to the appropriate power supply.

Feedback

Plausibility and Answer Analysis

The 'C' CCW pump can be powered from either the 6.9kV Bus 1A-SA or 1B-SB and a key interlock prevents racking in if 'A' or 'B' CCW pump is racked in on the same bus.

A. Correct.

B Incorrect. Plausible since the RHR, Containment Spray Pump and Chiller P-4 which are all safety related equipment, are powered from these buses. The key interlock prevents racking the breaker in if either 'A' or 'B' CCW pump is racked in on the bus.

C. Incorrect. Plausible since the power source is correct. The second part is plausible since the 'C' CSIP has a manual transfer switch which is allows for rapid pump swaps per OP-107 if required.

D. Incorrect. Plausible since the RHR, Containment Spray Pump and Chiller P-4 which are all safety related equipment, are powered from these buses. The second part is plausible since the 'C' CSIP has a manual transfer switch which is allows for rapid pump swaps per OP-107 if required.

Notes

008 Component Cooling Water

008K2.02 Knowledge of bus power supplies to the following: CCW pump, including emergency backup
(CFR: 41.7)

Importance Rating:	3.0* 3.2*
Technical Reference:	OP-145 Precaution and Limitation 10, Rev. 62
References to be provided:	None
Learning Objective:	CCW Objective 2.e
Question Origin:	NEW
Comments:	Must differentiate between 6.9 kV and 480 V emergency power supplies.
Tier/Group:	T2G1

37. 2011 NRC RO 037/BANK/FUNDAMENTAL/////008 K3.01/

Given the following plant conditions:

- The plant is operating at 100% power
- An inadvertent Containment Isolation Phase B has actuated

IAW AOP-018, Reactor Coolant Pump Abnormal Condition, which ONE of the following describes the operational impact on the RCPs if Phase B cannot be reset?

A. RCP operation may continue provided that RCP radial bearing temperatures remain less than 230°F

B. RCP operation may continue provided that normal seal injection flow is maintained

C✓ RCPs must be stopped within 10 minutes

D. RCPs must be stopped immediately

Feedback

Plausibility and Answer Analysis

Both AOP-018 and AOP-014 have attachments listing RCP trip limits. One trip limit is met when any RCP operates for ≥10 minutes without CCW flow to either motor oil cooler.

A. Incorrect. *Plausible since AOP-018 has the operator check RCP parameters and during an abnormal condition the operator would be monitoring the indications for trip limits but this trip limit would not be challanged with the isolation of CCW to the RCPs. The motor oil coolers would not have cooling flow and the motor bearing temperatures could be exceeded.*

B. Incorrect. *Plausible since the statement is correct but not for this situation. CCW flow to the RCP thermal barrier could be lost but as long as RCP seal injection flow is maintained the RCP can be ran indefinently.*

C. Correct.

D. Incorrect. *Plausible because PATH-1 step 10 with Containment pressure >10 psig would have the operator immediately trip the RCPs.*

Notes

008 Component Cooling Water

008K3.01 Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: Loads cooled by CCWS

Importance Rating:	3.4 3.5
Technical Reference:	AOP-018 RCP trip limites exceeded per Attachment 1 Page 28, Rev. 39
References to be provided:	None
Learning Objective:	CCWS Objective 9
Question Origin:	OIT Dev Bank CCWS-R1 3
Comments:	None
Tier/Group:	T2G1

38. 2011 NRC RO 038/BANK/FUNDAMENTAL/////010 K6.04/

The plant is operating at 100% power when the following sequence of events occurs:

- PZR Power Operated Relief Valve (PORV) 1RC-118 opens and sticks open.
- The associated PZR PORV Block valve, 1RC-117 cannot be closed.
- Pressurizer Relief Tank (PRT) pressure is increasing.

Which ONE of the following completes the statement below?

The PRT rupture disc will release prior to the PRT exceeding a design pressure of ____________ , after which point Pressurizer Power Operated Relief Valve tailpipe temperature will __________.

	PRT Rupture Disc Setpoint	PORV Tailpipe Temperature
A.✓	100 psig	Lower
B.	100 psig	Remain the same
C.	150 psig	Lower
D.	150 psig	Remain the same

Feedback

Plausibility and Answer Analysis

The PRT has 2 rupture discs. They are designed to rupture at 100 psig to prevent the PRT from exceeding it's design limit pressure of 100 psig. Per the TMI lessons learned (and basic thermodynamic principles), PORV outlet temperature will lower.

A. Correct

B. Incorrect. *Plausible, since the rupture disc blows at 100 psig. The applicant misapplies the constant enthalpy process and concludes that PORV outlet temperature remains the same.*

C. Incorrect. *Plausible, since the value is correct pressure for the Seal Water Return Line Relief valve setpoint (a relief valve that relieves to the PRT). Per the TMI lessons learned, and basic thermodynamic principles, PORV outlet temperature will lower.*

D. Incorrect. *Plausible, since the value is correct pressure for the Seal Water Return Line Relief valve setpoint (a relief valve that relieves to the PRT). The applicant misapplies the constant enthalpy process and concludes that PORV outlet temperature remains the same.*

Seal Water Return Line (IRC) Relief Valve, 1CS-467

This relief valve is on the seal water return header upstream of the inside containment seal water return isolation valve. This relief valve protects the piping and equipment in the seal water return line from overpressure. When the seal return line is isolated it will maintain a RCP seal leakoff flow path to the PRT. The relief valve lifts at 150 psig and it discharges to the PRT. It has sufficient capacity to pass the seal return flow from all three RCPs plus the excess letdown flow.

Notes

10 Pressurizer Pressure Control

010K6.04 Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: PRT
(CFR: 41.7 / 45.7)

Importance Rating:	2.9 3.2
Technical Reference:	FSAR Table 5.4.11-1
References to be provided:	None
Learning Objective:	PRZ Objective 2.e
Question Origin:	Bank
Comments:	None
Tier/Group:	T2G1

39. 2011 NRC RO 039/MODIFIED/FUNDAMENTAL/////012 A3.06/
With Reactor power stable at 45%, (1) what is the coincidence for the RCS Loop Lo Flow Reactor trip and (2) what is the basis for this Reactor trip?

A. (1) 2 of 3 bistables on 1 of 3 loops

(2) loss of fuel integrity

B. (1) 2 of 3 bistables on 2 of 3 loops

(2) loss of fuel integrity

C. (1) 2 of 3 bistables on 1 of 3 loops

(2) DNB protection

D✓ (1) 2 of 3 bistables on 2 of 3 loops

(2) DNB protection

Feedback

Plausibility and Answer Analysis

LOW PRIMARY COOLANT FLOW TRIP
This trip provides protection against DNB following a loss of flow condition. Loss of flow in a coolant loop is indicated by two out of three differential pressure flow signals less than the trip setpoint (90.5%). The following instruments provide input to the SSPS for their respective loops:

> *A loss of flow in a single coolant loop will cause a reactor trip if power is greater than P-8 (49% power). Above P-7 (10% power), but below P-8, a reactor trip will result from a loss of flow in two reactor coolant loops. Below P-7 this reactor trip function is automatically blocked.*

A. Incorrect — *Plausible since the Reactor trip single loop low flow only requires the loss on 1 of 3 loops. Second part is plausible because on a loss of a single pump the RCS temperature and fuel temperature will increase but not to the point of Fuel Pellet Melt and Fuel Pellet Melt is a reason for OP/Delta T in Tech Specs.*

B Incorrect — *First part is correct. Second Part is plausible since on a loss of a single RCP, the RCS temperature and fuel temperature will increase but not to the point of Fuel Pellet Melt and Fuel Pellet Melt is a reason for OP/Delta T in Tech Specs.*

C. Incorrect — *Plausible since the Reactor trip single loop low flow only requires the loss on 1 of 3 loops but this is above P-8 (49% power). Second part is correct*

D. Correct.

Notes

012 Reactor Protection

012A3.06 Ability to monitor automatic operation of the RPS, including: Trip logic (CFR: 41.7 / 45.5)

Importance Rating:	3.7 3.7
Technical Reference:	Tech Spec table 3.3-1 and RPS-LP-3.0/5.0
References to be provided:	None
Learning Objective:	RPS Objective 4.a
Question Origin:	OIT Dev Bank INTG 510
Comments:	None
Tier/Group:	T2G1

40. 2011 NRC RO 040/NEW/FUNDAMENTAL/////013 K1.18/

Given the following plant conditions
- The plant was operating at 100% power
- A LOCA has occurred and the crew is implementing PATH-1.
- The crew incorrectly resets the "SI Suction Auto Switchover" signal vice the "SI" signal when directed to reset SI in PATH-1.

Which ONE of the following completes the statement below?

The recirc sump suction valves to the __(1)__ will not automatically open when the RWST low-low level setpoint is reached and the CSIP alternate miniflow isolation valves __(2)__ automatically open and shut on RCS pressure.

A.✓ (1) RHR pumps ONLY

(2) will NOT

B. (1) RHR pumps and Containment Spray pumps

(2) will NOT

C. (1) RHR pumps ONLY

(2) will

D. (1) RHR pumps and Containment Spray pumps

(2) will

Feedback

Plausibility and Answer Analysis

The containment recirculation sump swapover (See Figure 9) functions to maintain a suction source to the RHR and Containment Spray pumps when the RWST inventory has reached the low-low level setpoint (end of the ECCS injection phase). The input relays for the actuation of this function are „energize to actuate“. When two of four RWST level instruments lower to 23.4% the following occurs:

> *With a SI actuation signal is present, the containment sump suction valves to the RHR system automatically open (1SI-300, 1SI-301, 1SI-310, 1SI-311). The operator will shut the RHR suction valves from the RWST (1SI-322, 1SI-323) per EOP-EPP-010, Transfer to Cold Leg Recirculation.*
>
> *If the containment spray pump is running (breaker closed), the containment sump suction valve for the containment spray train will automatically open*

(1CT-105, 1CT-102). When the sump suction valve is open a time delay relay is activated which will then send a signal to automatically shut the RWST suction valve for the containment spray train (1CT-26, 1CT-71). The time delay between the opening of the recirculation sump suction valve and closing of the RWST suction valve allows sufficient time for the sump suction line to fill. This assures an adequate available NPSH for the containment spray pump

The SI input signal for ECCS sump suction switchover does not reset with the SI reset switches. The SI Suction Auto Switchover Reset switches are used to reset the SI sump swapover function after SI has been reset. If it is reset early then the feature will not work when it should.

A. *Correct*

B. *Incorrect* *Part 1 is plausible since CS pump suction swapover could be mistaken to be initiated from the same signals as RHR pump suction swapover. The CS pump suction swapover actually is initiated by low-low RWST level and the breaker closed on the CS pump. Part 2 is correct.*

C. *Incorrect* *Part 1 is correct. Part 2 is plausible since actuation of the CSIP alternate miniflow automatic operation could be mistaken to be from the SI signal alone not the RWST-S signal and if that were the case, failure to reset SI would keep the signal functioning.*

D. *Incorrect* *Part 1 is plausible since CS pump suction swapover could be mistaken to be initiated from the same signals as RHR pump suction swapover. The CS pump suction swapover actually is initiated by low-low RWST level and the breaker closed on the CS pump. Part 2 is plausible since actuation of the CSIP alternate miniflow automatic operation could be mistaken to be from the SI signal alone not the RWST-S signal and if that were the case, failure to reset SI would keep the signal functioning.*

Notes

013 Engineered Safety Features Actuation System (ESFAS)

013K1.18 Knowledge of the physical connections and/or cause effect relationships between the ESFAS and the following systems: Premature reset of ESF actuation
(CFR: 41.2 to 41.9 / 45.7 to 45.8)

Importance Rating:	3.7 4.1
Technical Reference:	Logic Drawing, EPP-012 Step 4 Page 4 Rev. 26
References to be provided:	None
Learning Objective:	ESFAS Objective 8.j
Question Origin:	NEW
Comments:	None
Tier/Group:	T2G1

41. 2011 NRC RO 041/NEW/C/A/////022 A4.05/
Which ONE of the following identifies the earliest time that Tech Spec 3.6.1.5, Containment Systems Air Temperature, is required to be entered?

	TIME	HIGHEST Cnmt Air Temp	AVERAGE Cnmt Air Temp
A.	1000	122°F	118°F
B.	1010	121°F	116°F
C.	1020	124°F	119°F
D.✓	1030	126°F	121°F

Feedback

Plausibility and Answer Analysis

Per Tech Spec 3.6.1.5 in Modes 1, 2, 3, and 4. Primary Containment average air temperature shall not exceed 120°F. If the average air temperature exceeds 120°F TS 3.6.1.5 entry is required.

A. Incorrect — *Plausible since 118°F is the setpoint for the "Containment Average Temperature" annunciator and it would be conservative to use a peak/highest temperature. Incorrect because this is not the correct TS limit and not the correct parameter for evaluating the Containment temperature.*

B. Incorrect — *Plausible since 120°F is the TS limit and it would be conservative to use a peak/highest temperature. Incorrect because this is not the correct parameter for evaluating the Containment temperature per the TS.*

C. Incorrect — *Plausible for same reasons in A and B and since 118°F is the setpoint for the "Containment Average Temperature" annunciator. Incorrect because this is not the correct TS limit.*

D. Correct — *Average Containment Air Temperature has exceeded 120°F therefore TS 3.6.1.5 should be entered.*

Notes

022 Containment Cooling

022A4.05 Ability to manually operate and/or monitor in the control room: Containment readings of temperature, pressure, and humidity system
(CFR: 41.7 / 45.5 to 45.8)

Importance Rating:	3.8 3.8
Technical Reference:	TS 3.6.1.5
References to be provided:	None
Learning Objective:	Containment Cooling System Objective 11.a
Question Origin:	NEW
Comments:	None
Tier/Group:	T2G1

42. 2011 NRC RO 042/BANK/FUNDAMENTAL/////026 A4.01/

Which ONE of the following identifies the minimum required logic for MANUAL actuation of the Containment Spray System using the MCB Containment Spray activation switches?

A. Any ONE of the FOUR switches

B. Any TWO of the FOUR switches

C. Either the inside TWO switches OR the outside TWO switches

D✓ Either the left TWO switches OR the right TWO switches

Feedback

Plausibility and Answer Analysis

There are 4 channels for Containment Spray actuation logic on MCB. (2 channels per train) To manually initiate Containment Spray the operator must turn 2 switches on either the left panel or 2 on the right panel (BOTH switches from either train). In addition the Containment Spray logic is an energize to actuate circuit.

A. Incorrect. Plausible since each switch is labeled as CSAS, the operator may misinterpret the labeling and incorrectly believe 1 switch would be sufficient to actuate spray.

B. Incorrect. Plausible because it is partially correct. The reset does require operation of 2 of 4 switches, however they must be the 2 right or the 2 left switches in combination.

C. Incorrect. Plausible because it is partially correct. The reset does require operation of 2 of 4 switches, however they must be the 2 right or the 2 left switches in combination.

D. Correct.

Notes

026 Containment Spray

026A4.01 Ability to manually operate and/or monitor in the control room: CSS controls (CFR: 41.7 / 45.5 to 45.8)

Importance Rating:	4.5 4.3
Technical Reference:	Logic Drawing EMDRAC 1364-000871, CSS Actuation logic
References to be provided:	None
Learning Objective:	Containment Spray System Objective 4
Question Origin:	OIT Exam Bank CSS (04) 2
Comments:	None
Tier/Group:	T2G1

43. 2011 NRC RO 043/MODIFIED/C/A/////039 K5.08/

Given the following plant conditions:

- A Unit startup is in progress following a mid-cycle outage
- The Reactor is critical at 10^{-8} amps
- A SG PORV valve failed open

Which ONE of the following describes the initial response of SG level after the PORV opens and final Reactor power?
(Assuming NO action by the operating crew)

	SG Level	Reactor Power
A.	goes down	at the Point of Adding Heat
B.	goes down	exceed the POAH
C.	goes up	at the Point of Adding Heat
D.✓	goes up	exceed the POAH

Feedback

Plausibility and Answer Analysis

SG PORV's - Each valve is an electro-hydraulically operated valve with a capacity of 795,000 lbm/hr (5% - 6% rated steam flow per valve) when fully open and an inlet pressure of 1200 psig. If a SG PORV were to fail open with the Reactor critical then steam flow would increase causing RCS Tavg to lower with a corrosponding Reactor power increase. Without operator actions Reactor power will increase to approximately 5% - 6% which is exceeding the POAH (defined in GP-004, Reactor Startup) as 1% - 3% power.

A. Incorrect. *First Part is plausible because with an additional steam load of any SG there is a loss of inventory which ultimately will cause SG level to go down providing additional water is not added. Second Part is plausible because Reactor power is increasing therefore someone could believe that would cause RCS Tavg to also increase to do a rise in fuel temperature. But the steam leak will cause RCS temperature to lower; the size of the steam leak should approximate the final power level which in this case would be approxmimately 5% - 6% power making the second part of the answer (1% - 3% - at the POAH) incorrect.*

B. Incorrect. *First Part is plausible because with an additional steam load of any SG there is a loss of inventory which ultimately will cause SG level to go down providing additional water is not added. Second part is correct.*

C. Incorrect. *First Part is correct. Second Part is plausible because Reactor power is increasing therefore someone could believe that would cause RCS Tavg to also increase to do a rise in fuel temperature. But the steam leak will cause RCS temperature to lower; the size of the steam leak should approximate the final power level which in this case would be approxmimately 5% - 6% power making the second part of the answer (1% - 3% - at the POAH) incorrect.*

D. Correct.

Notes

039 Main and Reheat Steam

039K5.08 Knowledge of the operational implications of the following concepts as the apply to the MRSS: Effect of steam removal on reactivity
(CFR: 441.5 / 45.7)

Importance Rating:	3.6 3.6
Technical Reference:	GP-004 Page 25, Rev 52 MSSS Student Text SG PORV rating (page 10) Rev. 2
References to be provided:	None
Learning Objective:	MSSS Objective 6.c
Question Origin:	Bank
Comments:	None
Tier/Group:	T2G1

44. 2011 NRC RO 044/NEW/C/A/////059 A2.01/

Given the following plant conditions:

- A plant startup is in progress with Reactor power at 8%
- The 'B' Main Feed Pump is in operation

The following events occur:

- 'B' Main Feed Pump trips on overcurrent
- 'A' Main Feed Pump automatically starts but trips on low oil pressure
- The crew has entered AOP-010, Feedwater Malfunctions
- Steam Generator Levels are as follows:
  'A' S/G 30% NR and decreasing
  'B' S/G 29% NR and decreasing
  'C' S/G 28% NR and decreasing

Which ONE of the following describes (1) the current status of the AFW system and (2) how will the plant be stabilized?

A.✓ (1) ONLY Motor Driven AFW pumps have auto started

(2) the Reactor must be manually tripped IAW AOP-010

B. (1) ONLY Motor Driven AFW pumps have auto started

(2) Reactor power must be reduced to maintain SG levels 52% - 62% using AFW flow

C. (1) Currently AFW pumps must be MANUALLY started

(2) the Reactor must be manually tripped IAW AOP-010

D. (1) Currently AFW pumps must be MANUALLY started

(2) Reactor power must be reduced to maintain SG levels 52% - 62% using AFW flow

Feedback

Plausibility and Answer Analysis

A. Correct.

B. Incorrect. Plauasible because part 1 is correct. However, per AOP-010 if any S/G level drops to 30% then a Reactor trip is required. If the candidate does not recognize that S/G levels are less than 30%, the next step in AOP-010 would be to control AFW flowand maintain 52-62%.

C. Incorrect. Plausible because part 2 is correct. However, in part 1 the AFW pump will start automatically on a trip of the last running MFP.

D. Incorrect. Plausible, if the operator does not recognize AFW should start automatically and does not realize that S/G levels have dropped to 30% then this would be correct.

Notes

059 Main Feedwater

059A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Feedwater actuation of AFW system
(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating: 3.4* 3.6*

Technical Reference: AOP-010 Step 7 RNO page 6, Rev. 33

References to be provided: None

Learning Objective: Aux Feedwater Objective 7.a

Question Origin: NEW

Comments: None

Tier/Group: T2G1

45. 2011 NRC RO 045/NEW/C/A/////059 A3.06/

Given the following plant conditions:

- The plant is operating at 100% power
- A feed flow instrument on the 'C' SG causes the Main Feed Regulating Valve to go full open.
- 'C' SG level reaches 80%
- An automatic Reactor and Turbine trip occurs.

Current plant conditions:

- The 'A' Reactor Trip Breaker failed to open
- RCS temperature is 553°F

For the above listed conditions, what are the positions of the following feedwater components for the 'C' SG?

	1FW-191 (Main FW C Regulator)	1FW-217 (Main FW C Isolation)
A.	Open	Closed
B.	Open	Open
C.	Closed	Open
D.✓	Closed	Closed

Feedback

Plausibility and Answer Analysis

MAIN FEEDWATER ISOLATION

The main feedwater isolation signal is provided to isolate the main feedwater flowpath to the steam generators. The isolation of main feedwater is required to mitigate the effects of several events. The MFIS aids in limiting RCS cooldown on a steam line break. It will also limit the containment pressure rise for secondary breaks within the containment. The FWIS can help prevent SG overfill, which can cause RCS overcooling or potential damage to steamline valves and the main turbine. SG overfill is a concern when responding to a steam generator tube rupture (SGTR). The main feedwater isolation signal (MFIS) is actuated from any SI signal or a two of four high-high SG levels (P-14 - 78%). The MFIS closes the main feedwater isolation valves (1FW-159, 1FW-277, FW-217), the main feed regulating valves (FRVs, 1FW-133, 1FW-249, 1FW-191), the FRV bypass valves (1FW-140, 1FW-256, 1FW-198), and trips the turbine and main feedwater pumps.

A. *Incorrect.* *Plausible since this answer would be correct if the actuation of MFI on SG high level is not identified to go to the FRV and because the FRV closure at <564°F and P-4 will not isolate the FRV with the absence of P-4.*

B. *Incorrect.* *Plausible since this answer would be correct if the MFI signal on SG high level is not identified and because the FRV closure at <564°F and P-4 will not isolate the FRV with the absence of P-4.*

C. *Incorrect.* *Plausible since this answer would be correct if the actuation of MFI on SG high level is not identified and the failure of the FRV closure at <564°F and P-4 with the absence of P-4 is not identified.*

D. *Correct.* *The MFI actuation on SG high water level will close both the FRV and FWIV even though the FRV closure on <564°F and P-4 will not occur.*

Notes

059 Main Feedwater

059A3.06 Ability to monitor automatic operation of the MFW, including: Feedwater Isolation
(CFR: 41.7 / 45.5)

Importance Rating:	3.2* 3.3
Technical Reference:	OMM-004 Attachment 6, Page 51, Rev. 34
References to be provided:	None
Learning Objective:	ESFAS Objective 8.f
Question Origin:	NEW
Comments:	None
Tier/Group:	T2G1

NOTE: Gerald Laska supplied K/A replacement for this question on 2/1/2011
Original K/A 059A3.06 Ability to monitor automatic operation of the MFW, including: Turbine driven feed pump

Reason: HNP does not have Turbine driven Main Feedwater Pumps. Both the 'A' and 'B' Main Feedwater Pumps are Motor driven Pumps.

46. 2011 NRC RO 046/BANK/FUNDAMENTAL/////061 K5.01/

Given the following plant conditions;

- The plant experienced a Reactor trip from 100% power.
- The Operators have not operated any controls post-trip.
- The crew completed PATH-1, Reactor Trip and Safety Injection, and has entered EPP-004, Reactor Trip Response.
- Pressurizer level is 25% and slowly decreasing.
- All Steam Generator levels are between 12% and 18% narrow range and slowly rising.
- Steam Generators pressures are approximately 990 psig and slowly decreasing.
- Tavg is 545°F and slowly decreasing.
- RCS pressure is 2020 psig and slowly decreasing.

Which ONE of the following actions should be the first priority of the crew IAW with EPP-004 to address the conditions?

A. Establish Emergency Boration

B. Close MSIVs and bypass valves

C✓ Throttle Auxiliary Feedwater Flow

D. Initiate Safety Injection and Return to PATH-1

Feedback

Plausibility and distractor analysis

With the present SG Water Levels, AFW has auto started with full flow established. This AFW flow is excessive, producing excessive heat transfer between the SGs and RCS. This is resulting in RCS cooldown. EPP-004 will control temperature based on Table 1. Reducing AFW flow prior to closing the MSIVs.

A. Incorrect. Plausible because RCS Temperature is lowering which adds positive reactivity and will require boration to maintain required SDM

B. Incorrect. Plausible since closing the MSIVs will assist in controlling cooldown but temperature should be controlled using steam and feed flow prior to closing the MSIVs. This action is also included in EPP-004.

C. Correct.

D. Incorrect. Plausible because the Safety Injection Foldout Criteria is applicable with Pressurizer level being one of the monitored parameter. And with RCS Pressure is lowering, the automatic setpoint is being approached.

Notes

061 Auxiliary/Emergency Feedwater

061 K5.01 Knowledge of the operational implications of the following concepts as the apply to the AFW: Relationship between AFW flow and RCS heat transfer
(CFR: 41.5 / 45.7)

Importance Rating:	3.6 3.9
Technical Reference:	EPP-004 Table 1 Revision 21
References to be provided:	None
Learning Objective:	Auxiliary Feedwater Objective 4
Question Origin:	Bank OIT Dev. 061 K5.01 2
Comments:	None
Tier/Group:	T2G1

47. 2011 NRC RO 047/NEW/C/A/////061 A2.04/

Given the following plant conditions:

- The plant is operating at 100% power
- The 'A' MDAFW pump is under clearance
- A loss of B-SB 125V DC power occurs

Which ONE of the following identifies:

1.) The response of the TDAFW Pump after the pump start signal occurs
2.) The method of controlling TDAFW speed with the Trip and Throttle Valve (T&TV) IAW OP-137, Auxiliary Feedwater System?

A. (1) start and trip on overspeed

(2) throttle using controls at the local control panel

B.✓ (1) start and trip on overspeed

(2) throttle using the manual handwheel

C. (1) start and idle at 1000 rpm

(2) throttle using controls at the local control panel

D. (1) start and idle at 1000 rpm

(2) throttle using the manual handwheel

Feedback

Plausibility and Answer Analysis

B-SB DC powers governor control, the local panel, and the T&TV. The governor valve fails to full fuel without DC power causing the pump to OS and the local panel and T&TV motor have no power requiring the local handwheel.

A. Incorrect. Plausible since the TDAFW pump will OS and there are controls on the local panel for the T&TV.

B. Correct.

C. Incorrect. Plausible since there is a failure mechanism that would cause TDAFW to idle at 1000 rpm (ramp generator failure to minimum) and there are controls on the local panel for the T&TV.

D. Incorrect. Plausible since there is a failure mechanism that would cause TDAFW to idle at 1000 rpm (ramp generator failure to minimum) and manual handwheel use is required.

Notes

061 Auxiliary/Emergency Feedwater

061A2.04 Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: pump failure or improper operation
(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	3.4 3.8
Technical Reference:	AOP-025-BD page 3, Rev. 9 OP-137, Section 8.7 page 50, Rev. 30
References to be provided:	None
Learning Objective:	Auxiliary Feedwater Objective 11
Question Origin:	NEW
Comments:	None
Tier/Group:	T2G1

48. 2011 NRC RO 048/BANK/FUNDAMENTAL////EARLY SUBMITTAL/062 K3.03/

Given the following:

- The unit is operating at 100% power
- A Fire in Aux Bus 1E1 occurs, resulting in the loss of Aux Bus E

Which ONE of the following describes the battery charger(s) that temporarily lose power and will automatically be re-energized?

A. 1A-SA and 1B-SA

B. 1A-SB and 1B-SB

C. 1A-SA or 1B-SA

D✓ 1A-SB or 1B-SB

Feedback

Plausibility and Answer Analysis

Item directly evaluates a loss of AC power to the DC distribution system by evaluating the knowledge of the power supply to the safety rated battery chargers. The safety rated battery chargers are powered from seperate 480v MCC's off of each 6.9kv safety bus. A-SA and B-SB 6.9kv safety busses are supplied from Aux busses D and E, respectively. A loss of Aux bus E will result in a loss of the B-SB safety bus until it is re-energized by its EDG.

A. Incorrect. *Plausible because the loss of Aux bus E will cause a loss of a safety bus but will ONLY result in the loss of B-SB safety battery chargers which does not include the 1A-SA battery charger.*

B. Incorrect. *Plausible because the loss of Aux bus E will cause a loss of a safety bus but will ONLY result in the loss of B-SB safety battery chargers which does not include the 1A-SB battery charger.*

C. Incorrect. *Plausible because the loss of Aux bus E will cause a loss of a safety bus but will ONLY result in the loss of B-SB safety battery chargers which does not include the 1A-SA battery charger.*

D. Correct.

Notes

062 A.C. Electrical Distribution

062K3.03 Knowledge of the effect that a loss or malfunction of the ac distribution system will have on the following: DC system
(CFR: 41.7 / 45.6)

Importance Rating:	3.1 4.2
Technical Reference:	OP-156.01, Rev. 31, Page 101, 105, 107
References to be provided:	None
Learning Objective:	DC Power Objective 9
Question Origin:	2008 North Anna Bank
Comments:	None
Tier/Group:	T2G1

49. 2011 NRC RO 049/BANK/C/A/////063 A1.01/

Given the following plant conditions:

- A loss of AC power has occurred
- 125 VDC battery 1A-SA is currently loaded at rated load

DC load shedding has been performed to reduce the battery load to half of rated load.

How long will the battery be available to supply the remaining loads?

A. More than 8 hours

B. More than 6 hours but less than 8 hours

C. More than 4 hours but less than 6 hours

D. up to 4 hours

Feedback

Plausibility and Answer Analysis

Reducing the discharge rate on a battery increases the battery capacity in a non-linear function such that decreasing the discharge rate by half, increases the capacity by more than double.

A. Correct.

B. Incorrect. Plausible since the discharge rate has been halved, so it would appear that the capacity would be doubled, but it is a non-linear relationship.

C. Incorrect. Plausible since the discharge rate has been halved, so it would appear that the capacity would be doubled, but it is a non-linear relationship.

D. Incorrect. Plausible since the battery is rated for 4 hours, but at a discharge halving the discharge rate would increase the capacity.

Notes

063 DC Electrical Distribution

063A1.01Ability to predict and/or monitor changes in parameters associated with operating the DC electrical system controls including: Battery capacity as it is affected by discharge rate
(CFR: 41.5 / 45.5)

Importance Rating:	2.5 3.3
Technical Reference:	GFES - No Reference supplied
References to be provided:	None
Learning Objective:	DC Power Objective 8
Question Origin:	Bank OIT Exam Bank DCP (02A) 1
Comments:	None
Tier/Group:	T2G1

50. 2011 NRC RO 050/BANK/FUNDAMENTAL/////063 K1.02/
Which ONE of the following describes the normal power source for a safety-related 125-V DC bus?

480-V MCC through a...

A.✓ battery charger to the DC bus.

B. battery charger, through the DC battery, and then to the DC bus.

C. 7.5-KVA inverter, through the DC battery, and then to the DC bus.

D. 7.5-KVA inverter to the DC bus.

Feedback

Plausibility and Answer Analysis

A. Correct. 125VDC Bus is supplied directly by battery chargers

B. Incorrect. Plausible because the battery charger supplies the DC battery,

C. Incorrect. Plausible because the 7.5-KVA inverter has a DC power supply and is safety related equipment.

D. Incorrect. Plausible because the 7.5-KVA inverter has a DC power supply and is safety related equipment.

Notes

063 D.C. Electrical Distribution

063 K1.02 Knowledge of the physical connections and/or cause-effect relationships between the DC electrical system and the following systems: AC electrical system (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Importance Rating:	2.7 3.2
Technical Reference:	CAR-2166-G-042S01, 250V DC, 125V DC, & 120V UPS One Line Diagram
References to be provided:	None
Learning Objective:	DC Power Objective 2.b
Question Origin:	Bank OIT Exam Bank DCP (02B) 1
Comments:	None
Tier/Group:	T2G1

51. 2011 NRC RO 051/BANK/FUNDAMENTAL/////064 G2.4.34/
During performance of AOP-004, Remote Shutdown, which ONE of the following describes the responsibility of the BOP operator?

A. Locally operate and control CSIPs for RCS Makeup

B. Monitor and control plant conditions from the Auxiliary Control Panel (ACP)

C. Operate plant equipment at the Auxiliary Transfer Panel (ATP) and Switchgear Rooms

D✓ Locally trip RCP breakers and start and control EDGs locally to provide safety-related power

Feedback

Plausibility and Answer Analysis

A. Incorrect. Plausible if candidate confuses responibilities of the BOP operator. A is performed by RAB operator.

B. Incorrect. Plausible if candidate confuses responibilities of the BOP operator. B is performed by CRS.

C. Incorrect. Plausible if candidate confuses responibilities of the BOP operator. D is performed by OAC.

D. Correct. Per AOP-004, whether there is a fire or no fire.

Notes

064 Emergency Diesel Generator

064 G2.4.34 Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.
(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	4.2 4.1
Technical Reference:	AOP-004, Pages 10 and 19, Rev. 55
References to be provided:	None
Learning Objective:	AOP-LP-3.4 Objective 7
Question Origin:	OIT Dev Bank G2.4.34 2
Comments:	None
Tier/Group:	T2G1

52. 2011 NRC RO 052/NEW/C/A/////073 K5.02/

Given the following plant conditions:

- An elevated reading of 10 mR/hour was observed on RM-3601, the Letdown HX Valve Gallery Radiation Monitor.
- Health Physics identified a hot spot in the letdown piping 20 feet from the RM-3601 detector.
- Operations flushed the hot spot and relocated it to 5 feet from the RM-3601 detector.

Based on current conditions, which ONE of the following identifies the expected radiation reading on RM-3601?

(Assume the hot spot in the pipe is a point source)

A. 20 mR/hr

B. 40 mR/hr

C. 80 mR/hr

D.✓ 160 mR/hr

Feedback

Plausibility and Answer Analysis

A. *Incorrect.* Plausible if the square root of the distances is taken, instead of squared as they should be (10mR/hr x $20^{1/2}$ ft = 20 mR/hr x $5^{1/2}$ ft).

B. *Incorrect.* Plausible if the distances are not squared as they should be (10mR/hr x 20 ft = 40 mR/hr x 5 ft).

C. *Incorrect.* Plausible if a mathematical error is made (value selected as a distracter due to the progression of other numbers in distracters).

D. *Correct.* Using the formula $I_1d_1^2=I_2d_2^2$, the intensity of the source at 5 feet is calculated to be 160 mRem/hr.

Notes

073 Process Radiation Monitoring

073K5.02 Knowledge of the operational implications as they apply to concepts as they apply to the PRM system: Radiation intensity changes with source distance
(CFR: 41.5 / 45.7)

Importance Rating:	2.5 3.0
Technical Reference:	GFES - No references supplied
References to be provided:	None
Learning Objective:	Radiation Monitor System Objective 6.a
Question Origin:	NEW
Comments:	None
Tier/Group:	T2G1

On 3/29/2011 I informed Gerald Laska that we could not create a question dealing with HNP Process Radiation Monitors associated with a source distance relationship with liquid or gaseous monitors.
Gerald said that we could write a question on an area monitor that monitors system piping such as Letdown Radiation monitoring. This type of monitor could be considered a process monitor.

53. 2011 NRC RO 053/BANK/C/A////EARLY SUBMITTAL/076 K4.06/

Given the following plant conditions:

- 'A' NSW pump is running in normal alignment with 'B' NSW pump secured

Which ONE of the following identifies the Service Water valve alignment two minutes after 'B' Emergency Service Water (ESW) pump is taken to START?

- 1SW-40, NSW SUPPLY to 'B' ESW header will be __(1)__
- 1SW-276, ESW TO NSW COMMON RETURN will be __(2)__

	(1)	(2)
A.	shut	shut
B.	open	open
C.✓	shut	open
D.	open	shut

Feedback

Plausibility and Answer Analysis

INTERLOCKS:
1SW-39, 40, 274, 275 shut on start of associated ESW pump
1SW-270, 271 interlocked with associated pump

A. Incorrect. *Plausible since 1SW-40 would close however 1SW-276 would remain open due to not being interlocked with a pump start unless transferrred to the ACP.*

B. Incorrect. *Plausible if misconception that 1SW-40 and 1SW-276 opened on a start of the 'B' ESW pump. This is backwards from actual interlock conditions.*

C. Correct. *1SW-40 is interlocked to shut on the start of the assoicated ESW pump. 1SW-276 would be normally open and not receive a shut signal upon the start of the 'B' ESW pump.*

D. Incorrect. *Plausible if misconception that 1SW-40 opened on a start of the 'B' ESW pump however 1SW-276 has no interlock to close on any ESW pump start unless aligned to the ACP.*

Notes

076 Service Water System (SWS)

076K4.06 Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following: Service water train separation
(CFR: 41/7)

Importance Rating:	2.8 3.2
Technical Reference:	HNP CWD 6-G-0425 S01 and S02 FSAR 9.2.1.3.1
References to be provided:	None
Learning Objective:	Service Water System Objective 3.b
Question Origin:	Bank OIT Exam Bank ESWS (03D) 1
Comments:	None
Tier/Group:	T2G1

54. 2011 NRC RO 054/NEW/C/A/////078 G2.1.19/

Given the following plant conditions:

- The plant startup is in progress with the Reactor critical at 10^{-8} amps
- PI-9751.1 MCB Instrument Air Header Pressure instrument is broken and a work request has been generated
- Annunciator ALB-002-8-1, Instrument Air Low Pressure has just gone into alarm
- At 0800, the crew implemented AOP-017, Loss of Instrument Air
- The BOP runs group display AOP-017 on ERFIS
- ERFIS Instrument Air pressure point PIA9751 trends the following:

Time	PIA9751
0815	88.5 psig
0830	74.6 psig
0850	57.4 psig
0910	33.8 psig

Which ONE of the following identifies the FIRST time the crew will be required to manually Trip the Reactor IAW AOP-017?

A. 0815

B. 0830

C. 0850

D✓ 0910

Feedback

Plausibility and Answer Analysis

When Instrument Air pressure decreases to < 35 psig AOP-017 step 2 RNO directs tripping the Reactor if the Reactor is critical.

A. Incorrect. *Plausible since 85 psig is identified in AOP-017 Attachment 7 as the point where RCS letdown valves start to go to mid-position. Also below 90 psig the Service Air valve automatically closes.*

B. Incorrect. *Plausible since 75 psig is the low pressure alarm setpoint.*

C. Incorrect. *Plausible because FW flow control valves auto close when IA pressure reaches 60 psig but because of the intial power level AFW will be in service. Main Feedwater is NOT in operation. Feedwater flow will be maintained to all SG's in this condition and ALL SG levels should be above 30%.*

D. Correct.

Notes

078 Instrument Air System (IAS)

078G2.1.19 Ability to use plant computers to evaluate system or component status.

Importance Rating:	3.1 3.1
Technical Reference:	AOP-017 Step 2 RNO, Rev. 32
References to be provided:	None
Learning Objective:	ERFIS Objective 9.a
Question Origin:	NEW
Comments:	None
Tier/Group:	T2G1

55. 2011 NRC RO 055/BANK/C/A/////103 K1.08/

Given the following plant conditions:

- A LOCA has occurred
- RCS pressure is 600 psig and stable
- Containment pressure is 11.5 psig and lowering
- SI has NOT been reset
- The OAC has placed the reset switches for Containment Isolation Phase A and Phase B to RESET

Which ONE of the following describes the status of Containment Isolation Phase A and Phase B signals?

	Phase A	Phase B
A.	will NOT reset	will reset
B.	will NOT reset	will NOT reset
C.✓	will reset	will reset
D.	will reset	will NOT reset

Feedback

Plausibility and Answer Analysis

A. Incorrect. *Plausible because SI has not been reset, and it is the initiating signal for CISA*

B. Incorrect. *Plausible because of a combination of B and C. Applicant may misunderstand reset circuitry of either or both*

C. Correct. *Both are physically capable of reset even though SI is not reset.*

D. Incorrect. *Plausible because Containment pressure is still above the High-3 setpoint.*

Notes

103 Containment

103 K1.08 Knowledge of the physical connections and/or cause effect relationships between the containment system and the following systems: SIS, including action of safety injection reset
(CFR: 41.2 to 41.9 / 45.7 to 45.8)

Importance Rating:	3.6 3.8
Technical Reference:	PATH-1; ESFAS text
References to be provided:	None
Learning Objective:	ESFAS Objectives 8.c and 8.d
Question Origin:	Bank OIT Dev. 103 K1.08 3
Comments:	None
Tier/Group:	T2G1

56. 2011 NRC RO 056/BANK/C/A/////002 G2.2.40/

The plant is in Mode 4.

The RCS has the following leak rates:

- Primary to secondary – SG 'A'	0.08 gpm
- Primary to secondary – SG 'B'	0.11 gpm
- Primary to secondary – SG 'C'	0.07 gpm
- Leakage by the PZR Safeties to PRT	5.40 gpm
- Total leakage from RCS	5.82 gpm

Which ONE of the following RCS Tech Spec leakage limits is being exceeded?

A.✓ Primary to Secondary Leakage

B. Unidentified Leakage

C. Pressure Boundary Leakage

D. Identified Leakage

Feedback

Plausibility and Answer Analysis

Tech Spec 3.4.6.2 - RCS operational leakage shall be limited to: 150 gallons per day primary-to-secondary leakage through any one steam generator. The leakage of 'B' SG exceeds the amount allowed in Tech Specs, in this case the total leakage is 158.4 gallons per day.

A. Correct.

B. Incorrect. Plausible, candidate can confuse identified leakage limit of 10 gpm with the limit for unidentified leakage at 1 gpm. In that case the candidate would believe that the identified leakage limit would have been exceeded.

C. Incorrect. Plausible, PZR safety valve leakage can be misinterpreted to be pressure boundary valve leakage.

D. Incorrect. Plausible, if candidate performs a math error while converting SG tube leakage to GPM and includes this as identified leakage this answer could be correct.

Notes

002 Reactor Coolant

002G2.2.40 Ability to apply Technical Specifications for a system.

Importance Rating:	3.4 3.7
Technical Reference:	TS 3.4.6.2
References to be provided:	None
Learning Objective:	TS-LP-2.0/3.0/5.0/8.0 Objective 1.e
Question Origin:	OIT Dev Bank HARRIS AUDIT RO 04 44
Comments:	None
Tier/Group:	T2G2

57. 2011 NRC RO 057/PREVIOUS/FUNDAMENTAL////2009A NRC RO/014 A4.01/

Given the following plant conditions:

- A Reactor startup is in progress IAW GP-004, Reactor Startup
- The OAC is currently withdrawing Control Bank 'C' rods

Which ONE of the following is the indicated rod height on the Control Bank 'C' Step Counters at which Control Bank 'D' rods are expected to begin withdrawing and what is the expected rod speed?

	Control Bank 'C' Step Counter Indication	Control Bank 'D' Rod Speed
A.	100 Steps	48 SPM
B.	100 Steps	64 SPM
C.✓	128 Steps	48 SPM
D.	128 Steps	64 SPM

Feedback

Plausibility and Answer Analysis

Bank Overlap starts at an indicated height of 128 steps. 48 SPM is Manual withdrawal speed for the Control Banks.

A Incorrect. Bank Overlap starts at an indicated height of 128 steps. 100 steps is the duration of Bank Overlap. 48 SPM is correct for Manual withdrawal of the Control Banks.

B Incorrect. Bank Overlap starts at an indicated height of 128 steps. 100 steps is the duration of Bank Overlap. 64 SPM is for the withdrawal of the Shutdown Banks.

C Correct.

D Incorrect. Bank Overlap starts at an indicated height of 128 steps. 64 SPM is for the withdrawal of the Shutdown Banks.

Notes

014 Rod Position Indication

014 A4.01 Ability to manually operate and/or monitor in the control room: Rod selection control
(CFR: 41.7 / 45.5 to 45.8)

Importance Rating: 3.3 3.1

Technical Reference: AOP-001 Rev 35, Attachment 3, page 44
OP-104 Rev. 30, Page 17

References to be provided: None

Learning Objective: RODCS Objective 5.a

Question Origin: Bank

Comments: **Previous** 2009A NRC RO question

58. 2011 NRC RO 058/BANK/C/A/////017 K6.01/

Given the following plant conditions:

- The crew is performing EPP-004, Reactor Trip Response
- Natural circulation verification is in progress.
- Core Exit Thermocouples G2 and K5 have failed due to open circuits.

With these open circuit failures, the input from these thermocouples to the Inadequate Core Cooling Monitor (ICCM) will be __(1)__ and the subcooling margin calculated by ICCM will be __(2)__.

A.✓ (1) failed low

(2) unaffected

B. (1) failed low

(2) higher

C. (1) failed high

(2) unaffected

D. (1) failed high

(2) lower

Feedback

Plausibility and Answer Analysis

When a thermocouple fails or is taken out of service, a reading of 50°F will be displayed for that thermocouple on the MCR RVLIS display screen. A saturation temperature is calculated based on PRC9445. This saturation temperature is then compared with the average of the 5 hottest of all 51 operable thermocouples (Point TRC9300) or the average of RCS wide range TH temperatures, TE-413/423/433 (Point TRC9413), used only if TRC9300 is not available.

A. Correct. A failed low input will not be used on ICCM (Only highest temperature and lowest pressure).

B. Incorrect. Plausible if a failed low input is used in the ICCM calculation.

C. Incorrect. Plausible because RTDs fail high on an open circuit.

D. Incorrect. Plausible because RTDs fail high on an open circuit.

Notes

017 In-core Temperature Monitor

017K6.01 Knowledge of the effect of a loss or malfunction of the following ITM system components: Sensors and detectors
(CFR: 41.7 / 45.7)

Importance Rating: 2.7 3.0

Technical Reference: ICCM Leson plan

References to be provided: None

Learning Objective: ICCM Objective 5.d

Question Origin: Bank OIT Dev. 017 K6.01 1

Comments: None

Tier/Group: T2G2

59. 2011 NRC RO 059/NEW/FUNDAMENTAL/////027 K2.01/

Which ONE is the power supply for S-1A and S-1B, Containment Airborne Radioactivity Removal (ARR) Fans?

	S-1A	S-1B
A.	MCC 1A21-SA	MCC 1B21-SB
B.	480V Bus 1A1	480V Bus 1B1
C.	480V Bus 1D2	480V Bus 1E2
D.✓	MCC 1D11	MCC 1E11

Feedback

Plausibility and Answer Analysis

A. Incorrect. Plausible since fans are powered from 480V MCCs and could be mistaken to be safety-related components.

B. Incorrect. Plausible since fans are powered from 480V nonsafety-related power supplies and could be mistaken to be 480V bus powered vice 480V MCC powered.

C. Incorrect. Plausible since fans could be mistaken to be safety-related components and could be mistaken to be 480V bus powered vice 480V MCC powered.

D. Correct.

Notes

027 Containment Iodine Removal

027K2.01 Knowledge of bus power supplies to the following: Fans
(CFR: 41.7)

Importance Rating:	3.1* 3.4*
Technical Reference:	OP-168, Page 22, Attachment 1, Rev 32
References to be provided:	None
Learning Objective:	CVS Objective 5.a
Question Origin:	NEW
Comments:	None
Tier/Group:	T2G2

60. 2011 NRC RO 060/BANK/C/A/////029 A3.01/

Given the following:

- The unit is in Mode 6.
- Refueling activities are in progress.
- A fuel assembly has been slightly damaged during removal from the core.
- Radiation levels are rising steadily and are currently as follows:

  - REM-01LT-3502A-SA, Cnmt RCS Leak Detection Monitor, is in HIGH ALARM

Which ONE of the following describes the system isolation signal(s), if any, that have been automatically initiated as a result of this alarm?

A.✓ ONLY Normal Containment Purge.

B. ONLY Containment Pre-Entry Purge.

C. BOTH Normal Containment Purge AND Containment Pre-Entry Purge.

D. NEITHER Normal Containment Purge NOR Containment Pre-Entry Purge.

Feedback

Plausibility and Answer Analysis

HI-RADIATION ALARM (RM-3502A AND RM-3502B)
If a HIGH ALARM occurs on RM-3502A, it will cause any running Containment Normal Purge Supply Fan (AH-82A or AH-82B) to trip and the isolation valves for the 8" ventilation ducts to and from containment to shut (1CP-5, 1CP-9, 1CP-3 and 1CP-6).
If a HIGH ALARM occurs on RM-3502B, it will cause any running Containment Pre-Entry Purge Exhaust Fans (E-5A or E-5B) to trip, the isolation valves for the 42" ventilation ducts to and from containment to shut (1CP-4, 1CP-10, 1CP-1, 1CP-7) and CP-D50 to shut. The Containment Pre-Entry Purge Supply Fans (AH-81A and AH-81B) will trip because E-5A and E-5B are secured.
The RM-3502A/B High Alarm signals are not included in the Containment Ventilation Isolation signal path because they are not processed by SSPS. These signals cause the fans to trip by directly impacting their start/stop circuitry, rather than being passed through SSPS. The valves go shut based on the fans being secured. This is the basis for RM-3502A/B signals not being included in the CVI discussion.

A. Correct.

B. Incorrect. Plausible since it is easy to confuse normal and pre-entry purge.

C. Incorrect. Plausible if candidate confuses with 3561 monitors which will generate an isolation signal to both Normal and Pre-entry purge.

D. Incorrect. Plausible if recognition that 3502A generates an isolation signal.

Notes

029 Containment Purge

029A3.01Ability to monitor automatic operation of the Containment Purge System including: CPS isolation
(CFR: 41.7 / 45.5)

Importance Rating:	3.8 4.0
Technical Reference:	AOP-013, page 12, Rev. 14
References to be provided:	None
Learning Objective:	CVS Objective 6.b
Question Origin:	OIT Dev Bank 034 A4.01-1
Comments:	None
Tier/Group:	T2G2

61. 2011 NRC RO 061/NEW/FUNDAMENTAL/RW/OST-1817///034 K4.03/
Which ONE of the following completes the statement below?

During refueling operations the fuel assemblies are removed from the core using the Manipulator crane __(1)__ hoist that is equipped with a primary overload cutout limit switch that is set at a maximum of __(2)__ pounds.

A.✓ (1) main

(2) 2700

B. (1) main

(2) 3000

C. (1) auxiliary

(2) 2700

D. (1) auxiliary

(2) 3000

Feedback

Plausibility and Answer Analysis

A. Correct. *The Refueling Machine is used for moving fuel assemblies and the automatic overload limit is less than or equal to 2700 pounds.*

B. Incorrect. *Plausible the Refueling Machine is used for moving fuel assemblies but 3000 pounds is the 111% of the actual setpoint.*

C. Incorrect. *Plausible since Manipulator Crane has an Auxiliary Hoist which is used for latching and unlatching drive rods. The load of 2700 pounds is correct.*

D. Incorrect. *Plausible since Manipulator Crane has an Auxiliary Hoist which is used for latching and unlatching drive rods. 3000 pounds is the 111% of the actual setpoint.*

Notes

034 Fuel Handling Equipment

034K4.03 Knowledge of design feature(s) and/or interlock(s) which provide for the following: Overload protection

Importance Rating:	2.6 3.3
Technical Reference:	OST-1817 Rev. 10, and PLP-114 Attachment 2 Rev. 21
References to be provided:	None
Learning Objective:	Fuel Handling and Storage Objective 6
Question Origin:	NEW
Comments:	None
Tier/Group:	T2G2

62. 2011 NRC RO 062/BANK/FUNDAMENTAL/////071 A1.06/

Given the following plant conditions:

- A Waste Gas release is in progress when the associated radiation monitor exceeds its high alarm setpoint.

Which ONE of the following describes: (1) how the release will be automatically terminated AND (2) in which stack this radiation will be detected.

A. (1) 3WG-230, Gas Decay Tanks to Plant Vent Isolation Valve, CLOSES

(2) Stack 5

B✓ (1) 3WG-229, Waste Gas Decay Tanks E & F to Plant Vent, CLOSES

(2) Stack 5

C. (1) 3WG-230, Gas Decay Tanks to Plant Vent Isolation Valve, CLOSES

(2) Stack 5A

D. (1) 3WG-229, Waste Gas Decay Tanks E & F to Plant Vent, CLOSES

(2) Stack 5A

Feedback

Plausibility and Answer Analysis

A. Incorrect. Plausible because 3WG-230 is a release valve, but it is the manually operated isolation for 3WG-229. Second Part is correct.

B. Correct. Because IAW AOP-005-BD, 3WG-229 closes on a high alarm signal from REM-1WV-3546, Stack 5 PIG.

C. Incorrect. Plausible because 3WG-230 is a release valve, but it is the manually operated isolation for 3WG-229. Second part is plausible because the release goes through Stack 5, not 5A.

D. Incorrect. First part is correct. Second part is plausible because the release goes through Stack 5, not 5A.

Notes

071 Waste Gas Disposal

071A1.06 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
(CFR: 41.5 / 45.5)

Importance Rating:	2.5 2.8
Technical Reference:	AOP-005-BD rev. 8, Page 3
References to be provided:	None
Learning Objective:	GWPS Objective 8
Question Origin:	Bank OIT Dev. 071 A1.06 2
Comments:	None
Tier/Group:	T2G2

63. 2011 NRC RO 063/BANK/C/A/////072 K3.02/

Given the following plant conditions:

- The plant is operating at 100% power
- Irradiated Fuel movement is being performed in the Spent Fuel Pool in preparation for Refueling
- Fuel Handling Building area radiation monitor RM-1*FR-3564A-SA, fails HIGH and is declared inoperable

Which ONE of the following completes the statements below?

Fuel Handling Building Emergency Exhaust Unit E-12 __(1)__
Fuel movement __(2)__

A.✓ (1) Receives an auto start signal

(2) may continue with no additional actions required.

B. (1) Receives an auto start signal

(2) must be immediately suspended

C. (1) Does NOT receive an auto start signal

(2) may continue with no additional actions required

D. (1) Does NOT receive an auto start signal

(2) must be immediately suspended

Feedback

Plausibility and Answer Analysis

FHB RM-23 MONITORS

There are four RM-23s per train with three detectors feeding each RM-23. There is a total of 24 detectors arranged such that always one group of detectors is not shielded by a crane or other object. The detectors are general area G-M tubes. Each RM-23 has three active channels (one per detector) labeled MR/HR. The only function button is C/S. These monitors automatically place FHB in emergency ventilation line up. The actuation logic is one of twenty-four because any monitor for any RM-23 isolates the entire building and starts the associated A or B emergency ventilation train. Logic is one of three for any RM-23.
Specific actions are: starts exhaust Fan E-12 and/or E-13 and the open emergency exhaust inlet valves. It stops the normal supply and exhaust fans in the FHB and shuts the FHB operating floor dampers.

TS 3.3.3.1 Fuel movement can be performed since only one of the 3 radiation monitors has been declared inoperable and there are still one or more of the detectors operable (2 of 3 are still operable). NO LCO actions are met.

A. Correct.

B. Incorrect. Plausible since E-12 does receive a start signal, but only one south monitor has failed. If both south trains had failed or indicated high then fuel movement must be immediately suspended.

C. Incorrect. Plausible since E-12 does receive a start signal, but only one south monitor has failed. If both south trains had failed or indicated high then fuel movement must be immediately suspended. Other ESF actuations require more than one channel to fail before receiving an automatic action

D. Incorrect. Plausible since other ESF actuations require more than one channel to fail before receiving an automatic action

E-12 will auto start on high radiation from 3564ASA. With other equipment operating normally, the opposite train EDG is available to supply the other FHB ventilation train. Therefore, no further actions are required. Control Room ventilation must be operable, but not the reason for the question asked here.

Notes

072 Area Radiation Monitoring

072K3.02 Knowledge of the effect that a loss or malfunction of the ARM system will have on the following: Fuel handling operations
(CFR: 41.7 / 45.6)

Importance Rating:	3.1 3.5
Technical Reference:	TS 3.3.3.1, 3.9.12
References to be provided:	None
Learning Objective:	RMS Objectives 6.a and 9.b
Question Origin:	OIT Dev Bank 034 K6.02 1
Comments:	None
Tier/Group:	T2G2

64. 2011 NRC RO 064/NEW/C/A/////075 A2.03/

Given the following plant conditions:

- The plant was operating at 54%
- 'C' CWP under clearance
- 'B' CWP tripped and Condenser vacuum has began degrading
- A manual Reactor/Turbine trip was initiated per AOP-012, Partial Loss of Condenser Vacuum
- The operating crew is stabilizing the plant

Which ONE of the following identifies both the value of Condenser vacuum the Reactor/Turbine should have been manually tripped at per AOP-012 AND the expected response of Condenser vacuum after the Turbine trip?

	Manual Trip on Vacuum	Condenser Vacuum
A.	7.5 inches Hg	Degrading
B.	7.5 inches Hg	Improving
C.	5.0 inches Hg	Degrading
D✓	5.0 inches Hg	Improving

Feedback

Plausibility and Answer Analysis

The value of vacuum a Reactor - Turbine trip is identified in AOP-012. When Turbine load is less than 60% the trip setpoint is 5.0 inHg absolute. Once the turbine is tripped the condenser heat load will be within the capability of 1 CWP and Condenser vacuum will improve as CW temps lower.

A. Incorrect. Plausible since the value of vacuum a reactor/turbine trip is initiated at in AOP-012 when greater than 60% turbine load is 7.5 inHg absolute and the toggle point may be mistaken. The connection between steam load, CW temps, and vacuum may be mistaken.

B. Incorrect. Plausible since the value of vacuum a reactor/turbine trip is initiated at in AOP-012 when greater than 60% turbine load is 7.5 inHg absolute and the toggle point may be mistaken. Plausible since the value of vacuum a reactor/turbine trip is initiated at in AOP-012 when greater than 60% turbine load is 7.5 inHg absolute and the toggle point may be mistaken.

C. Incorrect. Plausible since the value of vacuum a reactor/turbine trip is initiated at in AOP-012 when less than 60% turbine load is 5.0 inHg absolute. The connection between steam load, CW temps, and vacuum may be mistaken.

D. Correct.

Notes

075 Circulating Water

075A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the circulating water system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Safety features and relationship between condenser vacuum, turbine trip, and steam dump
(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	2.5 2.7*
Technical Reference:	AOP-012-BD, Rev. 11, Section 1.0
References to be provided:	None
Learning Objective:	AOP-LP-3.12 Objective 2.c
Question Origin:	NEW
Comments:	None
Tier/Group:	T2G2

65. 2011 NRC RO 065/NEW/FUNDAMENTAL/////086 K5.04/
Given the following conditions:
- You are in the Admin Building PABX Room
- The fire suppression system has actuated

Which ONE of the following describes (1) the fire extinguishing agent used in the Admin Building PABX Room AND (2) the hazard level to personnel in the room from the extinguishing agent?

A. (1) CO2

(2) life threatening

B. (1) CO2

(2) NOT life threatening

C.✓ (1) Halon

(2) NOT life threatening

D. (1) Halon

(2) life threatening

Feedback

Plausibility and Answer Analysis

Halon is the agent used to fight the electrical fire (fire type) and it extinguishes the fire by interupting the chemical reaction at approximately 5%-10% concentration not by displacing the oxygen and halon has a low toxicity.

A. Incorrect. Plausible since CO2 is a common gaseous extinquishing agent also used for fighting electrical fires and the hazard with CO2 is life threatening (asphyxiation).

B. Incorrect. Plausible since CO2 is a common gaseous extinquishing agent also used for fighting electrical fires and since CO2 is not toxic and is life threatening due to the displacement of oxygen.

C. Correct.

D. Incorrect. Plausible since Halon is the agent used to fight the electrical fire (fire type) and since other gaseous extinguishing agents extinquish the fire by displacing oxygen which creates a life threatening asphyxiation hazard.

Notes

086 Fire Protection

086K5.04 Knowledge of the operational implication of the following concepts as they apply to the Fire Protection System: Hazards to personnel as a result of fire type and methods of protection
(CFR: 41.5 / 45.7)

Importance Rating:	2.9 3.5
Technical Reference:	Fire Pre-Plan: SEC04-16-M-0306 Rev. 10, Page 63; OPT-3800, Rev. 6 page 3
References to be provided:	None
Learning Objective:	Fire Protection Objective 7
Question Origin:	NEW
Comments:	None
Tier/Group:	T2G2

66. 2011 NRC RO 066/NEW/C/A/////G 2.1.8/

Given the following plant conditions:
- The plant is in Mode 6
- An Emergency call has just been received from the Fuel Handling Building (FHB) that there was a loud banging noise coming from the lower levels in the FHB
- Fuel Handlers are waiting for an assembly to be placed in the upender on the Reactor side prior to transferring the assembly to the 'A' Fuel Pool
- Cavity and Spent Fuel Pool levels are observed to be decreasing

IAW AOP-031, Loss of Refueling Cavity Integrity, which ONE of the following coordinated activities will be directed by the operators in the control room?

Once all fuel assemblies are safely stored...

A.✓ Direct the Fuel Handling crew to move the Fuel Transfer Cart to the Fuel Handling Building side

Dispatch an operator to shut 1PP-427, Fuel Transfer Tube Gate Valve

B. Direct the Fuel Handling crew to move the Fuel Transfer Cart to the Fuel Handling Building side

Direct Maintenance to install and inflate Fuel Pool gates to the Unit 1&4 Transfer Canal

C. Direct the Fuel Handling crew to maintain the Fuel Transfer Cart on the Reactor side

Dispatch an operator to shut 1PP-427, Fuel Transfer Tube Gate Valve

D. Direct the Fuel Handling crew to maintain the Fuel Transfer Cart on the Reactor side

Direct Maintenance to install and inflate Fuel Pool gates to the Unit 1&4 Transfer Canal

Feedback

Plausibility and Answer Analysis

AOP-031 has the crew verify that the Transfer Cart is parked in the FHB. The cart is currently on the Reactor side and therefor the MCR crew would direct the Fuel Handling crew to move the Transfer Cart to the FHB side in order to allow 1PP-427 to be shut.

A Correct.

B Incorrect. *Plausible because the direction would be to have the Fuel handling crew move the Fuel Transfer cart to the FHB side. It is plausible to direct Maintenance to install and inflate the Fuel Pool gates because this is the direction that would be provided in AOP-041 for a loss of Fuel Pool Level but is not a direction from AOP-031.*

C. Incorrect. *Plausible because other Progress Energy nuclear sites (such as Robinson) have the Fuel Transfer Cart parked on the REACTOR side in order to close the Fuel Transfer Tube Gate Valve. The second part of the answer is correct.*

D. Incorrect. *Plausible because other Progress Energy nuclear sites (such as Robinson) have the Fuel Transfer Cart parked on the REACTOR side in order to close the Fuel Transfer Tube Gate Valve. It is plausible to direct Maintenance to install and inflate the Fuel Pool gates because this is the direction that would be provided in AOP-041 for a loss of Fuel Pool Level but is not a direction from AOP-031.*

Notes

2.1 Conduct of Operations

G2.1.8 Ability to coordinate personnel activities outside the control room.
(CFR: 41.10 / 45.5 / 45.12 / 45.13)

Importance Rating:	3.4 4.1
Technical Reference:	AOP-031, Page 35, Rev. 17
References to be provided:	None
Learning Objective:	AOP-LP-031 Objective 3
Question Origin:	NEW
Comments:	None
Tier/Group:	T3

67. 2011 NRC RO 067/NEW/FUNDAMENTAL/////G 2.1.26/

Per SAF-NGGC-2175, Electrical Safety and Arc Flash Protection, which ONE of the following describes:

1) What is the *minimum* voltage that requires a safety person?
2) What a designated electrical safety person is *allowed* to do while critical work is in progress?

A. (1) When working on or near exposed energized parts >150 Volts.

(2) Verify that a correct component is being manipulated.

B. (1) When working on or near exposed energized parts >150 Volts.

(2) Hold open doors or covers for others to perform work.

C✓ (1) When working on or near exposed energized parts >300 Volts.

(2) Verify that a correct component is being manipulated.

D. (1) When working on or near exposed energized parts >300 Volts.

(2) Hold open doors or covers for others to perform work.

Feedback

Plausibility and Answer Analysis

A. Incorrect. Part 1 is plausible since 150V is the value when protector gloves are not required to be used with rubber gloves per SAF-NGGC-2175. Part 2 is correct.

B. Incorrect. Part 1 is plausible since 150V is the value when protector gloves are not required to be used with rubber gloves per SAF-NGGC-2175. Part 2 is plausible since it is reasonable that this activity is not distractive and it is listed in SAF-NGGC-2175 (as an activity that cannot be performed).

C. Correct.

D. Incorrect. Part 1 is correct. Part 2 is plausible since it is reasonable that this activity is not distractive and it is listed in SAF-NGGC-2175 (as an activity that cannot be performed).

Notes

2.1 Conduct of Operations

G2.1.26 Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen). (CFR: 41.10 / 45.12)

Importance Rating:	3.4 3.6
Technical Reference:	SAF-NGGC-2175
References to be provided:	None
Learning Objective:	None
Question Origin:	NEW
Comments:	None
Tier/Group:	T3

68. 2011 NRC RO 068/PREVIOUS/FUNDAMENTAL////2009A NRC RO/G 2.2.20/

Given the following plant conditions:

- The plant is operating at 100% power
- Troubleshooting is in progress on the 1A-SA ESCW Chiller IAW AP-929, Troubleshooting Guide
- Leads must be lifted in the Control Circuit to support the troubleshooting
- The lead lift does not pose a risk of personnel injury or equipment damage

Which of the following identify acceptable methods for documenting the lead lift IAW AP-929?

1. Clearance Order (OPS-NGGC-1301, Equipment Clearance)
2. Verification Sign-Off Sheet (OPS-NGGC-1303, Independent Verification)
3. Component Manipulation Sign-Off Sheet (OPS-NGGC-1308, Plant Status Control)
4. Add to the Work Order Instructions at the time of lift (ADM-NGGC-0104, Work Management Process)

A. 1 and 2

B.✓ 2 and 3

C. 3 and 4

D. 1 and 4

Feedback

Plausibility and Answer Analysis

A Incorrect. *OPS-NGGC-1303 is correct as listed in P&L#7 of AP-929 but OPS-NGGC-1301 is not. OPS-NGGC-1301 does maintain configuration but is used for safety of personnel and equipment and should not be used for configuration alone.*

B Correct. *Both OPS-NGGC-1303 and* OPS-NGGC-1308 are listed i*n P&L#7 of AP-929*

C Incorrect. *OPS-NGGC-1308 is correct as listed in P&L#7 of AP-929 but adding to the Work Order Instructions at the time of lift is not. OPS-NGGC-1308 does allow use of a work order to maintain configuration but the components must be added to the work order during planning, not in the field.*

D Incorrect. *OPS-NGGC-1301 does maintain configuration but is used for safety of personnel and equipment and should not be used for configuration alone. OPS-NGGC-1308 does allow use of a work order to maintain configuration but the components must be added to the work order during planning, not in the field.*

Notes

2.2 Equipment Control

G2.2.20 Knowledge of the process for managing troubleshooting activities.
(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	2.6 3.8
Technical Reference:	AP-929 Rev. 16, page 15
References to be provided:	None
Learning Objective:	None
Question Origin:	BANK
Comments:	**Previous** 2009A NRC RO question
Tier/Group:	T3

69. 2011 NRC RO 069/MODIFIED/C/A/////G 2.2.42/

Given the following plant conditions:
- The Reactor is shutdown for a scheduled refueling outage
- An RCS cooldown is in progress IAW GP-007, Normal Plant Cooldown

The following information is a plot of the cooldown:

TIME	RCS Tcold
0830	516°F
0845	505°F
0900	489°F
0915	478°F
0930	465°F
0945	425°F
1000	386°F
1015	382°F
1030	364°F
1045	349°F
1100	336°F
1115	320°F
1130	309°F
1145	297°F

At which time was the Tech Spec RCS Cooldown rate limit *FIRST* exceeded?

A. 0930

B. 1000

C.✓ 1030

D. 1145

Feedback

Plausibility and Answer Analysis

A. *Incorrect.* *Plausible since 51°F in 1 hour is greater than 50°F required by GP-007 if Tc < 350F, and is the first hour 50°F cooldown is met.*

B. *Correct.* *This is 103°F in one hour.*

C. *Incorrect.* *Plausible since limits were exceeded at 1030 (101°F cooldown in 1 hour with a limit of 100F/ hr when RCS >350F), but limits were FIRST exceeded at 1000.*

D. *Incorrect.* *Plausible since limits were exceeded at 1145 (52°F cooldown in 1 hour with a limit of 50F/ hr when RCS <350F. RCS cooldown rate limit changed at time 1045), but limits were FIRST exceeded at 1000.*

Notes

2.2 Equipment Control

G2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications.
(CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)

Importance Rating: 3.9 4.6

Technical Reference: TS 3.4.9

References to be provided: None

Learning Objective: TS-LP 2.0/3.0/5.0/8.0 Objective 4

Question Origin: OIT Dev Bank G2.2.42 1

Comments: None

Tier/Group: T3

70. 2011 NRC RO 070/BANK/FUNDAMENTAL/////G 2.3.4/
Which ONE of the following identifies (1) the annual administrative limit for TEDE at HNP AND (2) the Emergency Exposure Limit for protecting valuable property IAW PEP-330, Radiological Consequences?

A. (1) 5 rem

(2) 25 rem

B. (1) 2 rem

(2) 25 rem

C. (1) 5 rem

(2) 10 rem

D✓ (1) 2 rem

(2) 10 rem

Feedback

Plausibility and Answer Analysis

A. Incorrect. *Plausible because 5 Rem is the 10 CFR 20 TEDE annual exposure limit. 25 rem is the PEP-330*, Radiological Consequences, *limit for lifesaving or protection of large population activities.*

B. Incorrect. *Plausible because the first part is correct. 25 rem is the PEP-330,* Radiological Consequences, *limit for lifesaving or protection of large population activities.*

C. Incorrect. *Plausible because the second half is correct. 5 Rem is the 10 CFR 20 TEDE annual exposure limit.*

D. Correct.

Notes

2.3 Radiation Control

G2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12 / 43.4 / 45.10)

Importance Rating:	3.2 3.7
Technical Reference:	PEP-330, REV. 9, Page 17
References to be provided:	None
Learning Objective:	None
Question Origin:	Bank G2.3.4 1
Comments:	None
Tier/Group:	T3

71. 2011 NRC RO 071/BANK/C/A/////G 2.3.13/

Given the following plant conditions:
- The plant is operating at 100% power
- ALB-10-4-5, RAD MONITOR SYSTEM TROUBLE has alarmed
- Both REM-01LT-3502ASA, CNMT RCS Leak Detection Monitor, AND RM-1CR-3561A-SA, CNMT Vent Isolation Monitor, are in High Alarm

Which ONE of the following identifies actions required IAW annunciator panel procedures AND AOP-005, Radiation Monitoring System?

A. Verify Containment Ventilation Isolation

Monitor PZR level and CNMT conditions

B. Verify Containment Ventilation Isolation

Verify proper equipment alignment using OMM-004, Post-Trip/Safeguards Actuation Review

C✓ Verify Containment Purge isolated

Monitor PZR level and CNMT conditions

D. Verify Containment Purge isolated

Verify proper equipment alignment using OMM-004, Post-Trip/Safeguards Actuation Review

Feedback

Plausibility and Answer Analysis

A Containment Purge Isolation signal is generated when either REM-01LT-3502ASA or REM-01LT-3502BSB is in high alarm. IAW AOP-005 the operator should verify normal containment purge is isolated and then monitor for indications of primary leakage.

A. Incorrect. *Plausible because the candidate may confuse a Containment Purge Isolation coincidence signal (1/1 leak detection monitors) with a Containment Ventilation Isolation coincidence signal (2/4 Cnmt Vent Isolation monitors). The second part of the answer is correct.*

B. Incorrect. *Plausible because the candidate may confuse a Containment Purge Isolation coincidence signal (1/1 leak detection monitors) with a Containment Ventilation Isolation coincidence signal (2/4 Cnmt Vent Isolation monitors). The second part would be correct if the Cnmt Vent Isolation signal had occurred.*

C. Correct.

D. Incorrect. *Plausible because the first part is correct but the second part is only performed if a Containment Ventilation Isolation has actuated. A Containment Ventilation Isolation has not occurred. It takes 2 of 4 Cnmt RCS Leak Detection Monitors in High Alarm to cause the isolation signal. Only one of four monitors are in High Alarm.*

Notes

2.3 Radiation Control

G2.3.13 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. (CFR: 41.12 / 43.4 / 45.9 / 45.10)

Importance Rating:	3.4 3.8
Technical Reference:	ALB-10-4-5, Rev. 25, AOP-005, Rev. 27
References to be provided:	None
Learning Objective:	RMS Objective 6
Question Origin:	OIT Dev Bank G2.3.13 2
Comments:	None
Tier/Group:	T3

72. 2011 NRC RO 072/BANK/FUNDAMENTAL/////G 2.3.14/

The crew is implementing PATH-2 due to a SGTR. The ruptured SG PORV controller setpoint has been adjusted to 88% (1145 psig).

This controller setpoint is designed to serve which of the following purposes?

A. Minimize RCS to SG ΔP and minimize challenges to the Code Safety valves.

B.✓ Minimize atmospheric releases and minimize challenges to the Code Safety valves.

C. Minimize RCS to SG ΔP and maintains ruptured S/G PORV available for cooldown.

D. Minimize atmospheric releases and maintains ruptured S/G PORV available for cooldown.

Feedback

Plausibility and Answer Analysis

A. *Incorrect. Plausible since raising the setpoint does minimize RCS to SG D/P, but this is not the purpose of this step.*

B. *Correct.*

C. *Incorrect. Plausible since raising the setpoint does minimize RCS to SG D/P and the goal of PATH-2 during a SGTR is to prevent atmospheric release of radioactive fluid.*

D. *Incorrect. Plausible since the goal of PATH-2 during a SGTR is to prevent atmospheric release of radioactive fluid.*

Notes

2.3 Radiation Control

G2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.
(CFR: 41.12 / 43.4 / 45.10)

Importance Rating:	3.4 3.8
Technical Reference:	PATH-2 Background Doc, E-3, Rev. 1C, Page 60
References to be provided:	None
Learning Objective:	EOP-LP-3.2 Objective 2.a
Question Origin:	OIT Dev Bank G2.3.14 1
Comments:	None
Tier/Group:	T3

73. 2011 NRC RO 073/BANK/C/A/////G 2.4.12/

Given the following plant conditions:

- A LOCA has occurred
- Both CSIPs are tripped
- A RED Path CSF is confirmed on Core Cooling
- A RED Path CSF is confirmed on Heat Sink
- The operating crew is performing FRP-C.1, Response To Inadequate Core Cooling
- The RWST LO-LO LEVEL alarm just annunciated

Which ONE of the following identifies when the crew must transition to EPP-010, TRANSFER TO COLD LEG RECIRCULATION?

A. After receiving ALB-04/2-5, REFUELING WATER STORAGE TANK EMPTY alarm

B. When directed by FRP-C.1 to return to the procedure and step in effect

C. As soon as cold leg recirculation capability is confirmed

D✓ Prior to completing any other actions in FRP-C.1

Feedback

Plausibility and Answer Analysis

Per FRP-C.1 Foldout.

A. Incorrect. Plausible because the APP for ALB-4/2-5 and a foldout critera for EPP-010 has actions for the operator to take associated with CSIP, RHR and CT Pumps.

B. Incorrect. Plausible since this is the typical transition point in the EOP network.

C. Incorrect. Plausible since this is a PATH-1 decision-point for the choice between going to EPP-10 and EPP-12.

D. Correct.

Notes

2.4 Emergency Procedures / Plan

G2.4.12 Knowledge of general operating crew responsibilities during emergency operations.
(CFR: 41.10 / 45.12)

Importance Rating:	4.0 4.3
Technical Reference:	FRP-C.1 Foldout, Page 3 Rev. 16 EOP Users Guide Section 5.1.7, Page 17, Rev. 30
References to be provided:	None
Learning Objective:	EOP-LP-3.19 Objective 1.g
Question Origin:	Bank
Comments:	None
Tier/Group:	T3

74. 2011 NRC RO 074/BANK/FUNDAMENTAL/////G 2.4.22/

Given the following plant conditions:
- The plant is operating at 100% power
- A loss of offsite power occurs
- Both EDGs fail to start
- The crew entered EPP-001, Loss of AC Power To 1A-SA and 1B-SB

Which ONE of the following statements provides the reason for this caution in EPP-001?

CAUTION
Critical Safety Function Status Trees should be monitored for information only. Function Restoration Procedures should **NOT** be implemented unless directed by this procedure.

A. During a loss of AC power the CSF status tree inputs are unreliable.

B. The steps of EOP-EPP-001 are a higher priority than any of the steps in the FRPs.

C✓ The FRPs were designed under the assumption that at least one safety bus was energized.

D. FRPs assume that a design basis accident has occurred and a loss of all AC power is not a design basis accident.

Feedback

Plausibility and Answer Analysis

In accordance with the EOP-EPP-001 Basis Document about this Caution statement; "The guideline has priority over all FRGs and is written to implicitly monitor and maintain critical safety functions. This priority is necessary since all FRGs are written on the premise that at least one ac emergency bus is energized."

A. Incorrect. Plausible since on a loss of all AC some components will lose power (such as radiation monitors) requiring some CSFSTs to be evaluated manually.

B. Incorrect. Plausible since the candidate may know EPP-001 has priority and be mistaken that it is due to importance similar to the prioritization of CSFSTs.

C. Correct.

D. Incorrect. Plausible since loss of all AC is beyond design basis.

Notes

2.4 Emergency Procedures / Plan

G2.4.22 Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.
(CFR: 41.7 / 41.10 / 43.5 / 45.12)

Importance Rating:	3.4 4.4
Technical Reference:	EPP-001 Background EOP Users-Guide, Page 15, Rev. 30
References to be provided:	None
Learning Objective:	EOP-LP-3.19 Objective 2.b
Question Origin:	OIT Dev Bank INTG 413
Comments:	None
Tier/Group:	T3

75. 2011 NRC RO 075/NEW/C/A/////G 2.4.45/

Given the following plant conditions:
- The plant is operating at 55% power
- Both Condensate and Feedwater Trains are in service
- Both Heater Drain Pumps are in service

A transient occurs and the following annunciators are received:
- ALB-13-7-3, TWO RODS AT BOTTOM
- ALB-14-4-1B, SG A STM > FW FLOW MISMATCH
- ALB-16-1-4, FW PUMP A/B O/C TRIP - GND OR BKR FAIL TO CLOSE (due to a trip of the 'A' MFW pump)
- ALB-19-2-2A, HTR DRN PUMP A O/C TRIP-GND

Which ONE of the following identifies the annunciator that requires an entry into the EOP network?

A.✓ TWO RODS AT BOTTOM

B. SG A STM > FW FLOW MISMATCH

C. FW PUMP A/B O/C TRIP - GND OR BKR FAIL TO CLOSE

D. HTR DRN PUMP A O/C TRIP-GND

Feedback

Plausibility and Answer Analysis

A. Correct. *Two Rods at Bottom are entry conditions for AOP-001 and IAW the Immediate Actions, the reactor must be tripped anytime two or more rods are dropped.*

B. Incorrect. *Steam Flow > Feedwater Flow would indiciate a lowering SG Water level and would met entry conditions for AOP-010 but the reactor trip would not be required unless Water level could not be maintained greater than 30%.*

C. Incorrect. *The Feed Pump trip would meet entry conditions for AOP-010 but the Immediate Actions only require the reactor trip when initial power is greater than 90%. A trip is directed further along in AOP-010, however this is not an immediate action.*

D. Incorrect. *The Heater Drain Pump trip is entry conditions for AOP-010 but the immediate actions requiring a reactor trip are associated with trip of a MFP.*

Notes

2.4 Emergency Procedures / Plan

G2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm.
(CFR: 41.10 / 43.5 / 45.3 / 45.12)

Importance Rating:	4.1 4.3
Technical Reference:	AOP-001 Page 4, Rev. 35; OMM-027 rev.1, pg 7
References to be provided:	None
Learning Objective:	AOP-LP-3.1 Objective 4
Question Origin:	NEW
Comments:	None
Tier/Group:	T3