RO EXAMINATION ANSWER KEY

SALEM 2023 NRC Exam (21-01 ILOT) **– (Rev. 1)**

1 **10-150** **Points: 1.00**

Given:

- Salem Unit 2 is at 25% reactor power.
- The RO reports the following flow indications on 21 RC loop:
  - Channel 1 – 89%
  - Channel 2 – 89%
  - Channel 3 – 92%

Which of the following describes the expected response?

A. Per the Alarm Response Procedure, manually trip the reactor, stop 21 RCP, and enter 2-EOP-TRIP-1, Reactor Trip or Safety Injection.

B. RC flows are degraded but above the RPS setpoint for low RCS flow; power operation may continue.

C. Automatic reactor trip based on 2/3 low flow channels on one RC loop with power greater than P-7; perform immediate actions per 2-EOP-TRIP-1, Reactor Trip or Safety Injection.

D. Automatic reactor trip based on 1/3 low flow channels on one RC loop with power greater than P-7; perform immediate actions per 2-EOP-TRIP-1, Reactor Trip or Safety Injection.

Answer: A

Explanation / Justification:

A. Correct. Two out of three channels less than or equal to 90% of design loop flow indicates degraded RCP. Since reactor power is above P-7 (10%) and below P-8 (36%), no automatic reactor trip will occur with a loss of only one RC loop low flows. However, per S2.OP-AR.ZZ-0004 for OHA D-28, RCP BKR OPEN/FLO LO, with a degraded RCP flow, the RCP must be stopped, the reactor manually tripped, and enter 2-EOP-TRIP-1 since Tech Specs does not allow for three loop operation in Mode 1.

B. Incorrect. Plausible if the operator fails to recall the RC low flow setpoint of less than or equal to 90% design flow and believes the RCP is only degraded and does not meet the requirements for stopping the RCP. Incorrect in that the RCP is below the 90% flow requirements and must be stopped and the reactor tripped.

C. Incorrect. Plausible if the operator believes that the conditions meets the automatic RPS reactor trip for low RC flow. Incorrect in that above P-7 but below P-8, one RC loop low flow does not cause an auto Rx Trip.

D. Incorrect. Plausible if the operator believes that the conditions meets the automatic RPS reactor trip for low RC flow. Incorrect in that above P-7 but below P-8, one RC loop low flow does not cause an auto Rx Trip.

Technical References:	Logic drawing 221054, simplified ESF-2, S2.OP-AR.ZZ-0004 (R29)
Proposed References to be provided:	None
Learning Objective:	NOS05RXPROT-15, Objective 12.a. State the setpoints, coincidence, blocks and permissives for all Reactor Trips and Safety Injections actuations
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	RO
Category/System:	10-Reactor Trip
K/A #:	EPE 7 EA2.08 Ability to determine and/or interpret the following as they apply to a Reactor Trip: RCS loop flow rates
Importance:	3.0
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the operator must assess and/or interpret the RC loop flow conditions, recall RPS trip logic for RC loop low flows, and determine how the plant or the operators should respond.

Additional Reference: OHA Window D-28

S2.S2.OP-AR.ZZ-0004(Q)

D-28 (continued)

3.1 **DETERMINE** cause of alarm from Annunciator CRT.

CRT POINT	DESCRIPTION
413	Reactor Coolant Loop 21 Low Flow Channel I
414	Reactor Coolant Loop 21 Low Flow Channel II
415	Reactor Coolant Loop 21 Low Flow Channel III
416	RCP 21 Breaker Open Channel I

3.2 IF cause of alarm is failed instrument,

THEN NOTIFY Maintenance Controls to remove failed instrument from service IAW S2.OP-SO.RPS-0007(Q), RCP Channel Trip/Restoration,

AND REFER to Technical Specifications for applicability.

3.3 IF alarm is due to intentionally stopping 21 RCP as part of normal plant operations,

THEN NO further action is required.

3.4 IF a Reactor trip occurs,

THEN GO TO 2-EOP-TRIP-1, Reactor Trip or Safety Injection.

3.5 IF Reactor Coolant System flow degradation exists,

THEN:

A. **TRIP** the Reactor.

B. **STOP** 21 RCP.

C. IF RCP shutdown was due to RCP Seal Leakoff flow greater than or equal to 6 gpm, THEN simultaneously **PERFORM** the following:

- Between 3-5 minutes after stopping 21 RCP, **CLOSE** 21CV104, SEAL LEAKOFF.
- GO TO 2-EOP-TRIP-1, Reactor Trip or Safety Injection.

D. GO TO 2-EOP-TRIP-1, Reactor Trip or Safety Injection.

2 **10-151** **Points: 1.00**

Given:

- A small break LOCA and a Loss of ALL Offsite Power occurred on Salem Unit 2

Current plant conditions are as follows:

- SG pressures are 1015 psig and stable
- RCS pressure is 1600 psig and slowly lowering
- Highest CET temperature is 590 °F and lowering
- RCS T-Hot temperatures are 554 °F and lowering
- RCS T-Cold temperatures are 548 °F and stable
- 2B 4KV Vital Bus is locked out on Bus Differential

What is the primary method for RCS heat removal for current plant conditions?

A. Forced RCS Cooling

B. Natural Circulation

C. Reflux Boiling

D. ECCS Injection and Break flow

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the operator may forget that a LOOP has occurred and the 4KV group busses are not energized to run the RCPs for forced flow cooling.

B. Correct. Forced cooling is not available due to Loss of All Offsite Power. Only one Charging Pump is running and no SI pumps based on RCS pressure above the shutoff head (1540 psig), therefore, ECCS injection flow and break flow are insufficient for RCS heat removal. Natural circulation conditions are met based on the following parameters: (1) subcooling > 0°F. RCS pressure at 1600 psig has a saturation temperature or approx. 606 °F. Hottest CET is 590 °F, therefore subcooling is approx. 606 – 590 = 16 °F of subcooling, (2) SG pressures are stable and lowering, (3) RCS T-Hots are stable or dropping, (4) CETs are lowering, and (5) RCS T-colds are at saturation temp for SG pressure. SG pressure is 1015 psig (pre-set pressure for MS110s) which is a saturated temp from the outlet of the SGs of approx. 548 °F. Therefore per 2-EOP-LOCA-2 step 19.1 conditions for Natural Circulation are met and will be the RCS heat removal method.

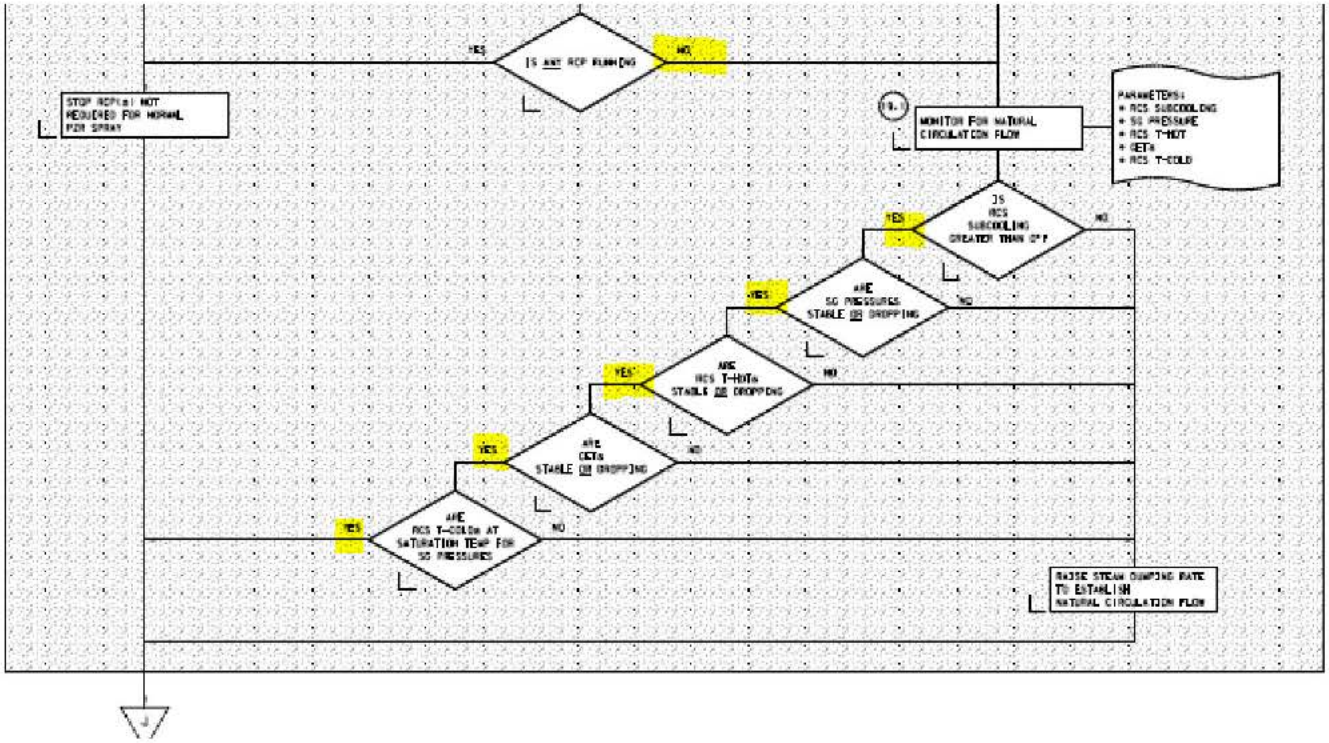
C. Incorrect. Plausible because Reflux cooling is a method for RCS cooling if none of the other methods are available.

D. Incorrect. Plausible because during some LOCA events, ECCS injection and break flow is the primary method for RCS heat removal (e.g., LBLOCA).

Technical References:	2-EOP-LOCA-2 (R42), ASME Steam Tables (year 2000)
Proposed References to be provided:	**ASME Steam Tables Rev. 2000**
Learning Objective:	NOS05LOCA02-06, Objective 5. Determine the indications that are monitored to ensure proper system/component operation for each step in the EOP for POST LOCA COOLDOWN AND DEPRESSURIZATION
Question Source:	Modified – Callaway 2017 NRC RO8
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.7 / 45.7 / 45.8
Level:	RO
Category/System:	10-Small Break LOCA
K/A #:	EPE 9 EK1.03 Knowledge of the operational implications and/or cause and effect relationship of the following concepts as they apply to Small Break LOCA: RCS heat removal
Importance:	4.0
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the applicant must assess plant conditions and determine the method of RCS heat removal during a small break LOCA.

Additional Reference:



Parent Question: Callaway 2017 NRC RO8

Modified question to change the answer to Natural Circulation is the RCS heat removal method.

Question # 8

A loss of off-site power and a small break LOCA occurred.

Current plant conditions are as follows:

- RCS pressure is 1800 psig and slowly lowering
  - SG pressures are 1135 psig and stable
- Highest Core Exit Thermocouple is 630°F and stable
- RVLIS Pumps Off indicates 60%
- RCS Hot Leg Temperatures are 562°F and stable
- RCS Cold Loop Temperatures are 530°F and stable
- NB01 is locked out
- 'B' CCP, tripped due to instantaneous overcurrent

What is the RCS Heat removal process(es) for the current plant conditions?

A. Forced RCS Circulation

B. Natural Circulation

C. Reflux Boiling

D. ECCS Injection Flow AND Small Break LOCA flow

Answer: C

3 **11-200** **Points: 1.00**

Given:

- Salem Unit 1 is experiencing a RCS leak inside containment.
- The crew is performing S1.OP-AB.RC-0001, Reactor Coolant System Leak.
- 11 charging pump is in service.
- The RO is attempting to stabilize PZR level by adjusting charging flow using the 1CV55, CHARGING FLOW CONTROL VALVE, and 1CV71, CHG HDR PCV.

Which of the following completes the below statement?

When 1CV71 is throttled…

A. open, seal injection rises and charging flow rises.

B. open, seal injection flow lowers and charging flow lowers.

C. closed, seal injection flow rises and charging flow lowers.

D. closed, seal injection lowers and charging flow rises.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because opening the 1CV71 will result in charging flow rising. Incorrect in that 1CV71 acts as backpressure control valve that affects both charging and seal injection flows. Opening the 1CV71 will cause backpressure in the charging line to lower resulting in an increase in charging flow but lower flow to the RCP seal injection lines.

B. Incorrect. Plausible because opening the 1CV71 will result in seal injection flow lowering. Incorrect in that 1CV71 acts as backpressure control valve that affects both charging and seal injection flows. Opening the 1CV71 will cause backpressure in the charging line to lower resulting in an increase in charging flow but lower flow to the RCP seal injection lines.

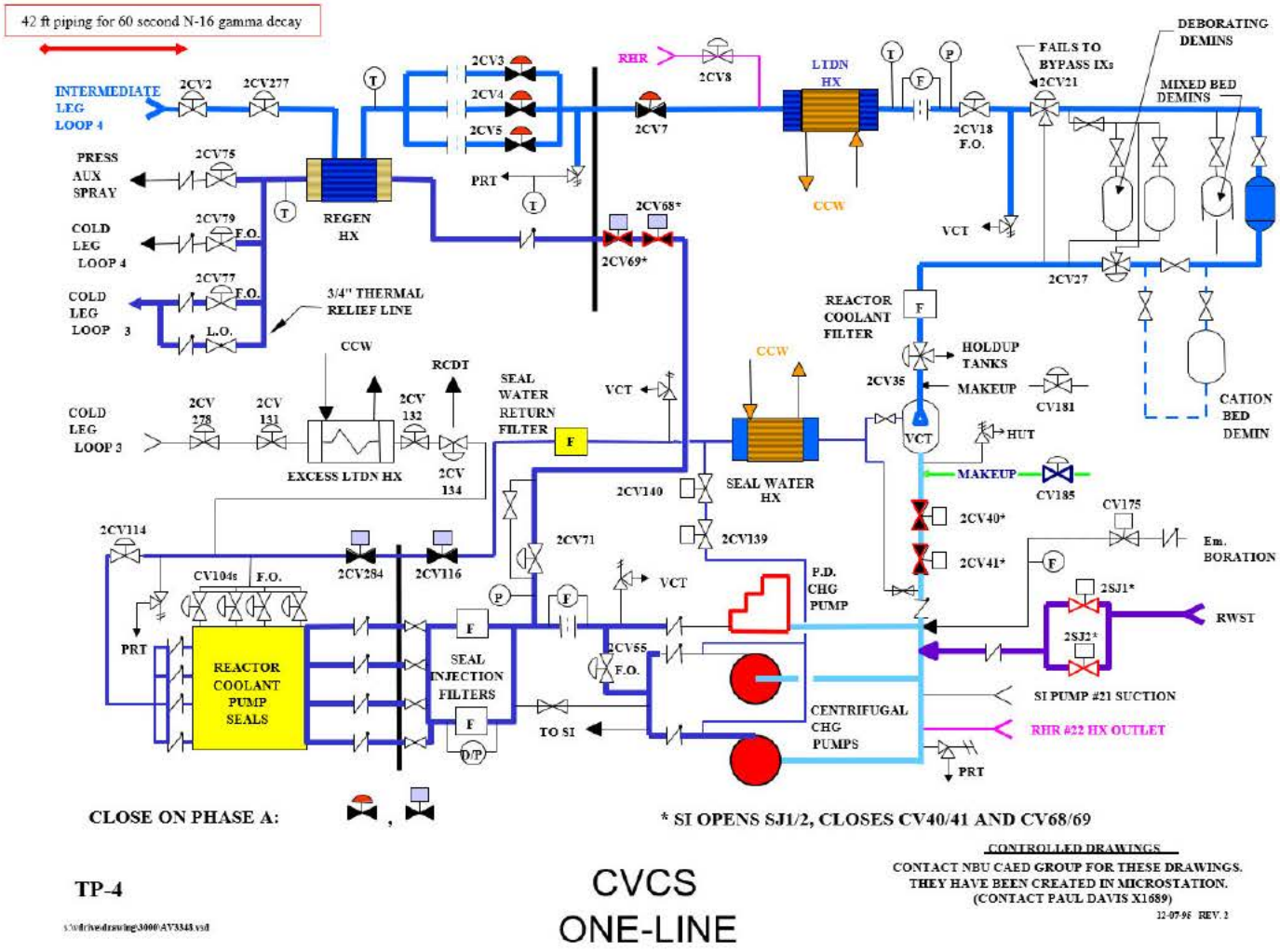
C. Correct. When the 1CV71 is closed, then backpressure to the seal injection lines will rise resulting in rising RCP seal injection flows and allowing less charging flow to the non-regen HX, thereby lowering charging flow to the RCS.

D. Incorrect. Plausible if the operator incorrectly recalls the location and operation of the 1CV55 and 1CV71 in the CVCS system and believes that the 1CV71 supplies flow the RCP seal injection only, thus if the 1CV71 is closed more flow will be diverted to charging flow and less to seal injection. . Incorrect in that 1CV71 open will cause seal injection to lower and charging flow to rise.

Technical References:	S1.OP-AB.RC-0001 (R18), CVCS one-line dwg
Proposed References to be provided:	None
Learning Objective:	NOS05CVCS00-18, Objective 4. Describe the function and how their normal and abnormal operations affects the Chemical and Volume Control System: Seal Pressure Control Valve, CV71.
Question Source:	Bank – Bryon 2017 NRC RO4
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.7 / 45.5 to 45.8
Level:	RO
Category/System:	11-Loss of Reactor Coolant Makeup
K/A #:	APE 22 AA1.05 Ability to operate and/or monitor the following as they apply to Loss of Reactor Coolant Makeup: RCP seal backpressure regulator valves.
Importance:	3.3
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the operator how operating the 1CV71 will effect charging flow and RCP seal injection flow during response to an RCS leak to stabilize PZR level.

Additional Reference: CVCS one-line



4 **11-201** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

The operating crew suspects an RCP Thermal Barrier rupture.

Which of the following indications would provide corroborating evidence for this condition, consider each choice separately?

1. 2R17A/B, Component Cooling Radiation Monitors, in Warning
2. PZR level rising
3. CCW Surge Tank level rising
4. Charging flow lowering
5. 2CC131, RCP Thermal Barrier Valve, goes closed

A. 1 and 3 Only

B. 2 and 4 Only

C. 2, 4, and 5 Only

D. 1, 3, and 5 Only

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because these indications will determine a Thermal Barrier leak. Incorrect in that the 2CC131 closing will also be an indication depending on the size of the leak.

B. Incorrect. Plausible because these would be indications if the operator believes that a Thermal Barrier leak will cause CCW to leak into the RCS system.

C. Incorrect. Plausible because these would be indications if the operator believes that a Thermal Barrier leak will cause CCW to leak into the RCS system.

D. Correct. A Thermal Barrier leak will cause CCW surge tank to rise as RCS will be leaking into the CCW system, RCS in-leakage will also be identified as a rise in the 2R17A or B radiation monitors that monitors activity in the CCW system, and the 2CC131 valve will also close on high CCW flow to the Thermal Barrier due to a thermal barrier leak, Therefore, choices 1, 3, and 5 are the correct indications for a Thermal Barrier leak into the CCW system.

Technical References:	S2.OP-AB.CC-0001 (R20)
Proposed References to be provided:	None
Learning Objective:	NOS05CCW000-16, Objective 4.c. Describe how the following components impact the Component Cooling Water System during normal and abnormal conditions:
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 45.12
Level:	RO
Category/System:	11-Loss of Component Cooling Water / Conduct of Operations
K/A #:	APE 26 G2.1.19 Ability to use available indications to evaluate system or component status.
Importance:	3.9
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the candidate is being ask to determine which indications in the control room will corroborate a thermal barrier leak in the CCW system.

5 **11-202** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- PZR Pressure controlling channel fails HIGH

Which of the following completes both statements concerning the effect from this failure?

Based on the conditions above, the __(1)__ reactor trip setpoint on the **unaffected** RCS loops will __(2)__.

A. (1) OTΔT (2) lower

B. (1) OTΔT (2) rise

C. (1) OPΔT (2) rise

D. (1) OPΔT (2) lower

Answer: A

Explanation / Justification:

A. Correct. The failed PZR pressure controlling channel will result in PZR pressure lowering. The lowering PZR pressure is one input into OTDT to determine its trip setpoint. As RCS pressure lowers, the OTDT setpoint will lower. With no operator action, the PZR pressure will continue to rapidly lower until the OTDT reactor trip setpoint is reached first (per S2.OP-AB.PZR-0001 bases section for Attachment 1).

B. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because the Low PZR Pressure Rx Trip is rate compensated as pressure lowers and will cause the Rx trip to occur at a higher pressure. Incorrect in that test runs in the simulator determined that OTDT will trip the reactor first.

C. Incorrect. Part 1 is incorrect. Plausible because the operator can confuse how PZR pressure affects the OTDT setpoint. Part 2 is correct.

D. Incorrect. Part 1 is incorrect. Plausible because the operator can confuse how PZR pressure affects the OTDT setpoint. Part 2 is incorrect. Plausible because the Low PZR Pressure Rx Trip is rate compensated as pressure lowers and will cause the Rx trip to occur at a higher pressure. Incorrect in that test runs in the simulator determined that OTDT will trip the reactor first

Technical References:	S2.OP-AB.PZR-0001 (R20)
Proposed References to be provided:	None
Learning Objective:	NOS05ABPZR1-05, Objective 2. Describe, in general terms, the actions taken in S2.OP-AB.PZR-0001 and the bases for the actions.
Question Source:	Bank – Salem
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	43.5 / 45.13
Level:	RO
Category/System:	11-Pressurizer Pressure Control System
K/A #:	APE 27 AA2.11 Ability to determine and/or interpret the following as they apply to a PZR pressure Control System Malfunction: RCS pressure
Importance:	4.0
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking to interpret a failed high PZR Pressure controlling channel and the effects on RCS pressure. The applicant needs to determine that the failure will cause RCS pressure to lower which will result in the OTDT setpoint to lower. The applicant is also asked which reactor trip will first based on lowering RCS pressure.

6 **10-152** **Points: 1.00**

Given:

- Salem Unit 2 is experiencing a Steam Generator Tube Rupture
- The crew is performing actions per 2-EOP-SGTR-1, Steam Generator Tube Rupture
- The CRS, RO, and BOP operators are the only licensed operators in the control room
- The RO needs to go to the Control Equipment Room (back rack area) to perform an action

Per OP-AA-101-111, Roles and Responsibilities of On-Shift Personnel, which of the following describes the requirements for this activity?

The RO…

A. must be relieved by a licensed operator other than the on-shift Unit 2 BOP operator.

B. may leave as long as the on-shift Unit 2 BOP operator remains “At the Controls” area.

C. may leave without being relieved by another licensed operator with CRS permission ONLY.

D. must be relieved by another licensed operator and a full turnover must be performed.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that during accident conditions two licensed operators must be in the control room or ATC at all times. Incorrect in that only one licensed RO is required to be in the ATC or inner-horseshoe during Modes 1-6.

B. Correct. Per OP-AA-101-111, at least one RO must be in the "At The Controls" (ATC) area (or inner-horseshoe) at all times during Modes 1-6, except for the time period required to acknowledge the receipt of an alarm outside the this area or to perform actions required for safe plant operation.

C. Incorrect. Plausible because the operator may believe SRO permission is required to leave the inner-horseshoe area. Incorrect in that only one licensed RO is required to be in the ATC or inner-horseshoe during Modes 1-6.

D. Incorrect. Plausible because the operator may believe that during accident conditions two licensed operators must be in the control room or ATC at all times. Incorrect in that only one licensed RO is required to be in the ATC or inner-horseshoe during Modes 1-6.

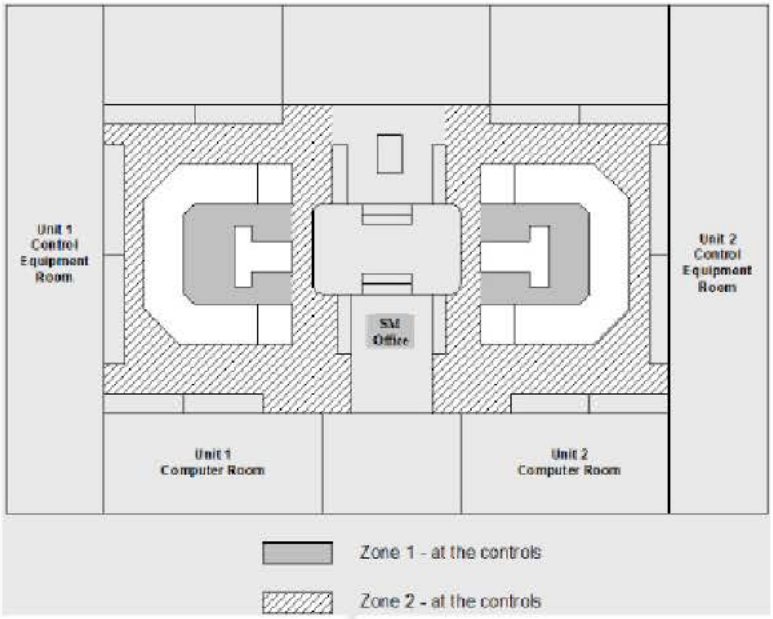
Technical References:	OP-AA-101-111 (R11)
Proposed References to be provided:	None
Learning Objective:	NOS05CONDOP-15, Objective 2.f and 2.g. Describe the position and purpose of the following positions in accordance with OP-AA-101-111, Roles and Responsibilities of On-Shift Personnel, and OP-AA-101-112, Roles and Responsibilities of Off-Shift Personnel: Reactor Operator (RO) and Plant Operator (PO)
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 45.13
Level:	RO
Category/System:	10-Steam Generator Tube Rupture / Conduct of Operations
K/A #:	EPE 38 G2.1.3 Knowledge of shift or short-term relief turnover practices.
Importance:	3.7
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the operator knowledge of short-term reliefs during an accident conditions.

Additional Reference: OP-AA-101-111

ROLES AND RESPONSIBILITIES OF ON SHIFT PERSONNEL **Rev: 11**

Attachment 4, Salem "At the Controls Area



Whenever the on-duty SM leaves the Control Area, the Unit 2 CRS assumes Control Room Command and Control unless the SM designates a different SRO.

- At least one RO is to be within Zone 1 at all times except for the time period required to acknowledge the receipt of an alarm outside this area or to perform actions required for safe plant operation.
  (Modes 1 - 6)
- SRO with command and control function may not enter the computer or equipment rooms.
- The on-duty SM and CRS(s) shall remain within the Protected Area.
- The STA shall be within 10 minutes of the Control Room Area in Modes 1 - 4.
- The STA shall be within 30 minutes of the Control Room Area in Modes 5 - 6, Defueled.

Modes 1 - 4 (per unit)

- SRO shall remain within the Control Room area. This SRO may not go in the opposite unit's computer and equipment rooms.

7 **10-153** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

Then:

- A reactor trip and Safety Injection occur due to the rapid depressurization of all steam generators.
- 2H and 2E 4KV Group Busses fail to auto transfer and are now de-energized
- The crew is performing actions in 2-EOP-LOSC-2, Uncontrolled Depressurization of All Steam Generators, to restore the 4KV Group Busses

Which of the following describes the current status of the PZR Backup Heaters?

A. ONLY 21 Backup Group is available, 22 Backup Group can be locally transferred to an emergency backup source.

B. ONLY 22 Backup Group is available, 21 Backup Group can be locally transferred to an emergency backup source.

C. NEITHER 21 or 22 Backup Groups are available, Backup Groups can be locally transferred to an emergency backup source.

D. BOTH 21 and 22 Backup Groups are available, Backup Groups will automatically transfer to an emergency backup source.

Answer: A

Explanation / Justification:

A. Correct. The preferred or normal power source for the PZR Heater Backup Groups is; Group 1 (21) – 2G and Group 2 (22) – 2E 4KV Group Busses. The PZR Heaters can be manually transferred to the emergency backup source; Group1 (21) from 2C and Group 2 (22) from 2A 460 V vital bus. EOP-LOSC-2 step 26 perform actions to restore the normal group busses or transfer PZR heater groups to the emergency backup power.

B. Incorrect. Plausible because the operator may incorrectly recall the normal power sources for the PZR Heaters.

C. Incorrect. Plausible because the operator may incorrectly recall the normal power sources for the PZR Heaters and believe that both PZR heaters are unavailable. Incorrect in that with 2H and 2E unavailable, PZR Heater Group 1 is available from 2G Group Bus.

D. Incorrect. Plausible because the operator may incorrectly recall the normal power sources for the PZR Heaters and believe that both PZR heaters are available. Incorrect in that with 2H and 2E unavailable, only PZR Heater Group 1 is available from 2G Group Bus. Incorrect in that the Heater Groups must be manually transferred to the vital 460V busses.

Technical References:	2-EOP-LOSC-2 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05PZRPL-12, Objective 5.b. State the power supply to the following Pressurizer Pressure and Level Control System components: Backup Heaters
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.8 / 41.10 / 45.3
Level:	RO
Category/System:	10-Steam Line Rupture
K/A #:	W E12 EK2.03 Knowledge of the relationship between Uncontrolled Depressurization of all Steam Generators and the following systems or components: AC electrical distribution system
Importance:	3.2
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the applicant about the normal AC power source to the PZR Heater backup groups and the availability of the emergency AC backup power source. EOP-LOSC-2 step 26 looks to restore the normal power source for the PZR heaters or transfer to the emergency backup source. Normal power is preferred to reduce the load on the EDGs,

8 **11-203** **Points: 1.00**

Given:

- Unit 2 is at 100% Power

Then:

- OHA G-22, FW HTR IN VLV TRIP & LVL HI, is Alarming
- PO reports high level in 21A Low Pressure Feedwater Heater

Which of the following describes the effect on plant operation?

1. Based on the above condition, __(1)__ CN22(s), Low Pressure FW Heater Inlet Valve, will be closed.
2. Actual reactor thermal power will be __(2)__.

A. (1) ALL (2) higher

B. (1) ALL (2) lower

C. (1) Only the 21 (2) lower

D. (1) Only the 21 (2) higher

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that any high level in one of the LP heater strings will result in isolation of all LP feedwater heater strings.

B. Incorrect. Plausible because the operator may incorrectly interpret how the isolation of the FW heater string will affect FW temperatures to the reactor and how the FW temperature change will affect reactor power and NIS indications.

C. Incorrect. Plausible because the operator may incorrectly interpret how the isolation of the FW heater string will affect FW temperatures to the reactor and how the FW temperature change will affect reactor power and NIS indications.

D. Correct. High water level in 21A Low Pressure feedwater heater will only close the associated 21CN22 inlet valve. Based on one LP FW heater string out of service, colder feedwater will enter the reactor. The cooler FW will have the effect of raising reactor thermal power and also provide some shielding to the PR NIS and result in the actual NI reactor power being lower than actual. As a result reactor power should be determined based on RCS ΔTs and turbine load should be reduced to prevent exceeding reactor thermal power limits.

Technical References:	S2.OP-AR.ZZ-0007 (R67)
Proposed References to be provided:	None
Learning Objective:	NOS05ABCN01-07, Objective 3. Describe, in general terms, the actions taken in AB.CN-0001 and the bases for the actions in accordance with the technical bases document.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 43.5 / 45.12
Level:	RO
Category/System:	11-Loss of Main Feedwater / Equipment Control
K/A #:	APE 54 G2.2.44 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.
Importance:	4.2
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the candidate is asked how the high level in a feedwater heater will affect the isolation of the three LP feedwater heater strings and how reactor power will be effected as a result of the isolated feedwater heater string.

9 **10-154** **Points: 1.00**

Which of the following describes the capacity of the station batteries during a station blackout?

Note: 2-EOP-LOPA-1, Loss of ALL AC Power

A. 2 hours assuming 2-EOP-LOPA-1 load shedding.

B. 2 hours during worst case loading (no 2-EOP-LOPA-1 load shed)

C. 6 hours during worst case loading (no 2-EOP-LOPA-1 load shed)

D. 6 hours assuming 2-EOP-LOPA-1 load shedding.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that the battery capacity of 2 hours is when load shedding is performed per 2-EOP-LOPA-1. Incorrect in that load shedding will extend the battery to 4 hours.

B. Correct. The 125 VDC vital batteries are designed to have adequate capacity to supply its Station Blackout (SBO) coping and restoration loads during the four (4) hour SBO coping duration with manual stripping of some non-essential loads (ref. section 7.2.2.3 in VTD 314204). The non-essential load sheds allow for extending vital battery availability. Salem FSAR Section 8.1.4.1 states that the batteries are designed for LOOP only, Loss of All AC Power Is not a Design Bases Accident (DBA). NOTE: the batteries are rated for two (2) hours of operation following a Loss of All AC Power without load shed.

C. Incorrect. Plausible if the operator may incorrectly recall and confuse the 4 hour extended battery capacity with the 6 hours.

D. Incorrect. Plausible if the operator may incorrectly recall and confuse the 4 hour extended battery capacity with the 6 hours. Incorrect in that load shedding will extend the battery to 4 hours.

Technical References:	S2.OP-AB.LOOP-0001 (R32), PSBP 314204 (SBO calculation), FSAR Sect 8.1.4.1
Proposed References to be provided:	None
Learning Objective:	NOS05DCELEC-11 Objective 2. Describe the design bases of the DC Electrical System.
Question Source:	Bank – Beaver Valley 2017 NRC RO11
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.5 / 41.10 / 45.6 / 45.13
Level:	RO
Category/System:	10-Station Blackout
K/A #:	EPE 55 K3.01 Knowledge of the reasons for the following responses and/or actions as they apply to Station Blackout: Length of time for which battery capacity is designed.
Importance:	4.1
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the applicant must have knowledge of the design capacity of the 125 VDC vital batteries during an SBO event.

10 **11-204** **Points: 1.00**

Given:

- Salem Unit 2 experienced a Loss of Off-Site Power
- The crew is performing S2.OP-AB.LOOP-0001, Loss of All Offsite Power
- ALL Charging Pumps on Unit 2 are unavailable
- 13 Charging Pump from Unit 1 is being aligned to supply charging flow to Unit 2

Then:

- Power to the 4KV Group Busses have been restored
- Preparations for re-starting RCPs are in progress
- RCP seal inlet temperatures are;
    - 21 RCP - 227 °F
    - 22 RCP - 220 °F
    - 23 RCP - 219 °F
    - 24 RCP - 221 °F

In accordance with S2.OP-AB.LOOP-0001, which of the following completes the statement concerning restoring RCP seal injection flow?

Based on the above conditions, restoring seal injection flow to the RCPs __(1)__ permitted. The reason for checking RCP seal inlet temperatures is to prevent ___(2)___.

A. (1) is NOT (2) actuation of the Safe Shutdown seal

B. (1) is (2) actuation of the Safe Shutdown seal

C. (1) is (2) damage to #1 seal

D. (1) is NOT (2) damage to #1 seal

Answer: D

Explanation / Justification:

A. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because the RCP seal package also includes the Safe Shutdown Seal which is a passive device that actuates on high temperature. RCP seal outlet temperature is monitored to limit seal outlet temperature below 190 °F, this is to prevent inadvertent actuation of the Safe Shutdown Seal (see S2.OP-AB.RCP-0001 bases). The operator may confuse that RCP seal inlet temperature is the limiting temperature for the Safe Shutdown Seal. Incorrect in that RCP seal outlet temperature is monitored to prevent Safe Shutdown Seal actuation.

B. Incorrect. Plausible because the operator may incorrectly recall the seal inlet temperature limitation and incorrectly recall when and how the Safe Shutdown Seal actuates.

C. Incorrect. Plausible because the operator may misinterpret the note in S2.OP-AB.LOOP-0001 Attachment 10 and believe that seal injection can be restored unless ALL RCP seal inlet temperatures are ≥ 225 °F.

D. Correct. Per S2.OP-AB.LOOP-0001, Attachment 10, if ANY one of the RCP seal inlet temperatures is ≥ 225 °F, then restoring seal injection flow to the RCPs is NOT permitted. Based on 21 RCP seal inlet temperature is ≥ 225 °F, then restoring seal injection flow to the RCPs is NOT permitted and in order to supply charging from 13 charging pump, Unit 2 must isolate seal injection to the RCPs prior to starting 13 charging pump. The reason for this is to prevent thermal shock from restoring seal injection and damage to the RCP seals.

Technical References:	S2.OP-AB.LOOP-0001 (R34)
Proposed References to be provided:	None
Learning Objective:	NOS05ABLOP1-09, Objective 2. Describe, in general terms, the actions taken in S2.OP-AB.LOOP-0001 and the bases for the actions
Question Source:	Bank - Salem
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.10 / 45.6 / 45.13
Level:	RO
Category/System:	11-Loss of Offsite Power
K/A #:	APE 56 AK3.02 Knowledge of the reasons for the following response and/or actions as they apply to Loss of Offsite Power: Actions contained in AOPs
Importance:	4.1
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the applicant the reason why seal injection is not restored based on seal inlet temperatures.

Additional Reference:

S2.OP-AB.LOOP-0001(Q)

ATTACHMENT 10
(Page 1 of 2)

PLACING 13 CHARGING PUMP INSERVICE
TO UNIT 2 FROM THE UNIT 1 RWST

NOTE

___ When utilizing 13 Charging Pump, Unit 1 should monitor RWST level to ensure compliance with Technical Specification 3.5.5. If Unit 1 RWST approaches the minimum volume required to satisfy Technical Specification 3.5.5, then either `continued operation of the 13 Charging Pump should be evaluated or appropriate compensatory actions implemented.

___ 1.0 IF **ANY** 21-24 RCP Seal Inlet Temperature is greater than or equal to 225°F, THEN **PERFORM** the following to establish charging flow ONLY to the RCS for inventory control:

___ ♦ **CLOSE** 2CV83, SEAL WATER FILTER INLET.

___ ♦ **CLOSE** 2CV89, SEAL WATER FILTER INLET.

___ ♦ **CLOSE** 2CV95, SEAL WATER FILTER BYPASS.

___ 2.0 **ENSURE** 13 Charging Pump is aligned for Appendix R Only (Unit 2) IAW S1.OP-SO.CVC-0002(Q), Charging Pump Operation.

Attachment 10 provides instructions for aligning 13 Charging Pump to Unit 1 when all Unit 2 Charging Pumps are unavailable. This attachment aligns 13 Charging Pump via the CVCS Cross-Connect System from the Unit 1 RWST to supply charging flow for RCS inventory control and boration capability to facilitate cooldown to Mode 5, and RCP seal injection after all RCP seal inlet temperatures are verified to be less than 225°F. If ANY 21-24 RCP Seal Inlet Temperature is greater than or equal to 225°F, then 13 Charging Pump is NOT started until ALL Unit 2 RCP seal injection isolation valves are closed to prevent thermal shock of the RCP seals when 13 Charging Pump is started.

11 **11-205** **Points: 1.00**

Given:

- Salem Unit 1 is at 8% power performing a reactor startup

Then:

- 1B Vital Instrument Bus de-energizes

Which of the following describes the resulting plant response?

A. The reactor does NOT trip because the Nuclear Instrumentation monitors will fail low.

B. The reactor does NOT trip because power is still below P-10.

C. A reactor trip occurs because one Intermediate Range channel bi-stable de-energizes.

D. A reactor trip occurs because one Source Range channel bi-stable de-energizes.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because the NI channels will fail low on loss of power, however, on loss of power the affected channel bi-stables will trip sending a signal to SSPS trip logic.

B. Incorrect. Plausible because the operator may believe that with reactor power above P-6 the SR and IT channels are blocked. Incorrect in that the IR high flux is blocked at P-10 (10%).

C. Correct. When the 1B vital instrument bus de-energizes the 2N32 (SR), 2N36 (IR) and 2N41 (PR) channels will lose instrument and control power and the monitors will fail low. In addition, on the loss of instrument power the affect NI channels will cause the bi-stables to SSPS to trip resulting in a trip signal to the SSPS trip logic. In this case with reactor power above P-6 (1E-5%) but below P-10 (10%), the reactor will trip on 1/2 IR channel high flux bi-stables tripped. The SR channels are blocked above P-6 and the PR reactor trip is based on 2/4 PR channels high flux. The IR high flux channels are block above P-10.

D. Incorrect. Plausible because the operator may incorrectly recall when the SR channels are blocked and believe that the loss of power to 2N32 will cause a reactor trip based on 1/2 SR channel bi-stables tripped. Incorrect in that the SR channels are blocked above P-6.

Technical References:	221052, 221053
Proposed References to be provided:	None
Learning Objective:	NOS05EXCORE-14, Objective 12.b. Given plant conditions, relate the Excore Nuclear Instrumentation System with the following: 115VAC Electrical System
Question Source:	Modified Bank - Byron Oct 2019 NRC RO58
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.8 / 41.10 / 45.3
Level:	RO
Category/System:	11-Loss of Vital AC Instrument Bus
K/A #:	APE 57 AK2.08 Knowledge of the relationship between Loss of Vital AC Instrument Bus and the following systems or components: NI.
Importance:	4.1
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the operator on how the loss of vital instrument power will affect the Nis. In this case the operator is being asked whether a reactor trip will occur when instrument power is lost requiring knowledge of reactor trip setpoints, coincidence, and permissives.

Parent Question:

Modified question based on the answer being different for Salem's design.

58 **ID: 2019 NRC-2 Q 58** **Points: 1.00**

Unit 1 is at 8% power during a plant startup.

Instrument bus 111 BYPASS SOURCE TO LOAD pushbutton is pressed.

What is the resulting plant response?

A. The reactor does NOT trip because power is rapidly transferred to the CVT.

B. The reactor does NOT trip because power is still below P-10.

C. A reactor trip occurs because N35 momentarily de-energizes.

D. A reactor trip occurs because N31 momentarily de-energizes.

Answer: A

12 **11-206** **Points: 1.00**

Given

- Salem Unit 2 experienced a loss of the 2A 125 VDC bus

Then:

- A Safety Injection actuation occurs

Based on the above conditions, which of the following completes the statement describing the remote operation of the 2A EDG and why?

The 2A EDG...

A. will NOT start because the EDG start circuit is supplied from DC power.

B. will NOT start because the SEC will not have power to energize the diesel start relay.

C. will start because SSPS directly starts the EDG on an SI and has redundant DC power supplies.

D. will start because the EDG start circuit is supplied from vital instrument power.

Answer: A

Explanation / Justification:

A. Correct. The EDG starting circuits are supplied from 125 VDC power. With no DC power, the EDG will not be able to start automatically from an SI/SEC signal or remotely from the control room.

B. Incorrect. Plausible because the operator may incorrectly recall that the SEC start relays are supplied from DC power. Incorrect in that the SEC relays are supplied by vital instrument power.

C. Incorrect. Plausible because SSPS does send a signal that would start the EDGs. Incorrect in that SSPS sends an SI signal to the SECs which the SECs will send a start signal to the EDG start circuitry which is DC powered.

D. Incorrect. Plausible because vital instrument power is used in many control circuits. Incorrect in that the EDG start circuit requires DC power.

Technical References:	NOS05EDG000-15
Proposed References to be provided:	None
Learning Objective:	NOS05EDG000-15, Objective 14.c. Given plant conditions, relate the Emergency Diesel Generator with the following: DC Distribution System.
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.5 / 41.10 / 45.6 / 45.13
Level:	RO
Category/System:	11-Loss of DC Power
K/A #:	APE 58 AK3.01 Knowledge of the reasons for the following responses and/or actions as they apply to Loss of DC Power: Operation of the EDGs.
Importance:	4.0
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the questions is asking the operator how the loss of DC power will affect the starting of the EDG and the reason why.

13 **11-207** **Points: 1.00**

Which of the following completes both statements concerning the design feature of the 2R13 radiation monitors?

1. The 2R13 radiation monitors are designed to monitor the radioactive effluent release pathway from the __(1)__.
2. A loss of the __(2)__ Circulating Water system flow will prevent monitoring of the effluent pathway.

A. (1) CFCUs
(2) Unit 1

B. (1) CFCUs
(2) Unit 2

C. (1) CCW Heat Exchangers
(2) Unit 2

D. (1) CCW Heat Exchangers
(2) Unit 1

Answer: A

Explanation / Justification:

A. Correct. The discharge of the CFCUs are monitored by the R13s for radioactivity from containment into the SW system. The CFCUs SW flow discharges into the opposite units Circulating Water system.

B. Incorrect. Plausible because the operator would expected the discharge of the CFCUs to go to its respective CW system.

C. Incorrect. Plausible because SW does flow through the CCW HXs and could believe that an effluent pathway may exist from the RHR HXs to SW system. Incorrect that this is not the design purpose of the R13s.

D. Incorrect. Plausible because SW does flow through the CCW HXs and could believe that an effluent pathway may exist from the RHR HXs to SW system. Incorrect that this is not the design purpose of the R13s.

Technical References:	S2.OP-SO.RM-0001, ODCM 3.3.8
Proposed References to be provided:	None
Learning Objective:	NOS05RMS000-20, Objective 3.i. Describe how the following components impact the Radiation Monitoring System during normal and abnormal conditions: R13A and B, CFCU Service Water Monitors
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.8 / 41.10 / 45.3
Level:	RO
Category/System:	11- Loss of Nuclear Service Water
K/A #:	APE 62 AK2.08 Knowledge of the relationship between Loss of Service Water and the following system or components: PRMS
Importance:	3.0
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the applicant how the loss of service water will affect the operability of the 2R13s radiation process monitors.

Additional References: S2.OP-SO.RM-0001 and ODCM 3.3.8

ACTION 28 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that:

a. At least once per 8 hours, local monitor readouts for the affected channels are verified to be below their alarm setpoints, or
b. With a Service Water System leak (inside containment) on the Containment Fan Coil Unit associated with the inoperable monitor either:
   1. At least once per 8 hours, grab samples are to be collected and analyzed for principal gamma emitters, I-131, and dissolved and entrained gases at the lower limits of detection specified in ODCM CONTROL Table 4.11-1.B, and the ODCM Surveillance Requirement 4.11.1.1.2 is performed, or
   2. Isolate the release pathway.
c. With no identified service water leakage (inside containment) on the Containment Fan Coil Unit associated with the inoperable monitor, at least once per 24 hours, collect grab samples and analyze for principal gamma emitters, I-131, and dissolved and entrained gases at the lower limits of detection specified in ODCM CONTROL Table 4.11-1.B, and the ODCM Surveillance Requirement 4.11.1.1.2 is performed.

14 **11-208** **Points: 1.00**

Given:

- Salem Unit 2 experienced reactor trip and Safety Injection due to an Inadvertent SI
- All systems were in normal alignment and in AUTO prior to the reactor trip
- Instrument air to containment is isolated
- PZR PORVs are cycling on the air accumulators to maintain PZR pressure

Then:

- Instrument Air to containment is restored

Which of the following describes the status of the PZR PORVs and spray valves when air to containment is restored?

PZR PORVs air accumulators ___(1)___ automatically align back its normal air supply header and the PZR spray valves ___(2)___ automatically re-open when air is restored.

A. (1) will
(2) will NOT

B. (1) will NOT
(2) will NOT

C. (1) will NOT
(2) will

D. (1) will
(2) will

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that on loss of air in containment that the PZR spray valves will have to be manually re-opened.

B. Incorrect. Plausible because the operator may believe that operator action is required in containment to realign the accumulators back to the normal air supply.

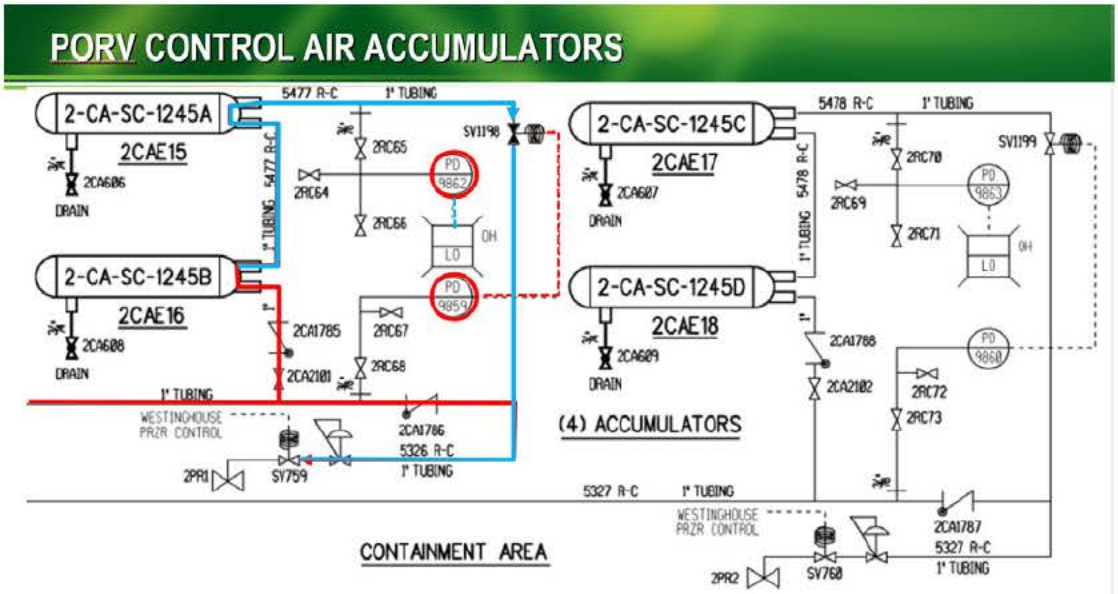
C. Incorrect. Plausible because the operator may believe that operator action is required in containment to realign the accumulators back to the normal air supply.

D. Correct. The PZR PORV air accumulators will automatically isolate and be supplied by the normal containment control air when air pressure in containment is restored and the Master Pressure Controller will have a demand to open spray valves, therefore when air to containment is restored both spray valves will re-open.

Technical References:	NOS05CONAIR-13
Proposed References to be provided:	None
Learning Objective:	NOS05CONAIR-13, Objective 4.b. Describe how each impact the Control Air System during normal and abnormal conditions: PORV Control Air Accumulators
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.7 / 45.5 to 45.8
Level:	RO
Category/System:	11- Loss of Instrument Air
K/A #:	APE 65 AA1.03 Ability to operate and/or monitor the following as they apply to Loss of Instrument Air: Restoration of systems served by instrument air when pressure is regained
Importance:	3.1
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the applicant how the PZR PORV control air accumulators realign following restoring control air to containment and whether the spray valves will re-open.

Additional Reference:



Normal Air is available - > 90 psig

Accumulator aligned - < 85 psig

15 **11-209** **Points: 1.00**

Given:

- Both Units 1 and 2 are at 100% power with Voltage Regulators in AUTO
- Salem Unit 1 is at 100% power with 350 MVARs OUT (lagging)
- Salem Unit 2 is at 100% power with 200 MVARs OUT (lagging)

Then:

- Salem Unit 1 trips

Based on the above condition, which of the following completes the statement describing the status of the 500 KV grid?

Grid voltage will __(1)__ and Salem Unit 2 MVARs will __(2)__.

A. (1) rise
(2) rise

B. (1) rise
(2) lower

C. (1) lower
(2) lower

D. (1) lower
(2) rise

Answer: D

Explanation / Justification:

A. Incorrect. First part is incorrect. Plausible because the operator may incorrectly believe that a unit trip will rise grid voltage. Second part is correct.

B. Incorrect. First part is incorrect. Plausible because the operator may incorrectly believe that a unit trip will rise grid voltage. Second part is incorrect. Plausible if the operator confuses the concept on how grid MVARs respond to grid voltage and believes that a lower grid voltage will cause a lowering of MVARs.

C. Incorrect. First part is correct. Second part is incorrect. Plausible if the operator confuses the concept on how grid MVARs respond to grid voltage and believes that a lower grid voltage will cause a lowering of MVARs.

D. Correct. Per S2.OP-AB.GRID-0001, Note for step 1.4, a loss of either Salem Unit 1 or 2 may result in the 500KV switchyard voltage lowering below 493 KV or by 2%. As the grid voltage lowers, the generating units on the grid will experience sharing of the reactive load and result in, in this case, Salem Unit 2 MVARs rising following the trip of Salem Unit 1.

Technical References:	S2.OP-AB.GRID-0001 (R23)
Proposed References to be provided:	None
Learning Objective:	NOS05ABGRID-13, Objectives 3.b. Given a set of initial plant conditions: Describe the plant response to actions taken in the abnormal procedure.
Question Source:	Modified – Salem 2022 NRC RO17
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	RO
Category/System:	11-Generator Voltage and Electric Grid Disturbances
K/A #:	APE 77 AA2.05 Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: Status of grid
Importance:	3.7
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the operator the effect on the grid following a trip or loss of a Salem unit. In addition, the operator is being asked how the MVARs (reactive load) will be affected following the loss of Salem Unit 1.

Parent Question: Salem 2022 NRC RO17

Modified Part 2 of the question to test what happens to grid voltage.

Given:

- Both Units 1 and 2 are at 100% power with Voltage Regulators in AUTO.
- Unit 1 reactive load is 250 MVARs OUT.
- Unit 2 reactive load is 250 MVARs OUT.
- The 500 KV Ring Bus is closed.

Subsequently, Unit 2 trips.

Which ONE of the following completes both statements?

1. Unit 1 Main Generator Reactive (MVAR) loading will __(1)__.

2. Generator field current will __(2)__.

A. 1) rise by less than 250 MVAR
   2) rise

B. 1) rise by less than 250 MVAR
   2) lower

C. 1) rise 250 MVAR
   2) rise

D. 1) rise 250 MVAR
   2) lower

16 **10-029** **Points: 1.00**

Given:

- Salem Unit 2 is responding to a Loss of Coolant Accident per 2-EOP-LOCA-6, LOCA Outside Containment.
- ALL ECCS Pumps are running
- The crew suspects a leak on an SI Pump discharge Cold Leg
- RCS pressure is 1700 psig and steady

Which of the following indications would corroborate the leak and the valve(s) required to be closed to isolate the leak per 2-EOP-LOCA-6?

A. SI Pump flows indicating 200 gpm; 2SJ135, Cold Leg Discharge Valve

B. SI Pump flows indicating 200 gpm; 21 and 22 SJ49, Cold Leg Isolation Valves

C. SI Pump pressure at 1700 psig; 21 and 22 SJ49, Cold Leg Isolation Valves

D. SI Pump pressure at 1700 psig; 2SJ135, Cold Leg Discharge Valve

Answer: A

Explanation / Justification:

A. Correct. An indicate SI Pump flow 200 gpm with RCS pressure at 1700 psig would indicate a leak on the SI Cold Leg. RCS pressure at 1700 psig is above the shutoff head of an SI Pump. Therefore, flow indicated on the SI Pumps indicates that a leak exists on the cold leg(s). Per 2-EOP-LOCA-6 step 2, the 2SJ135 Cold Leg Discharge Valve is closed to isolate a leak on either of the SI Pump Cold Legs.

B. Incorrect. Plausible because 21 and 22 SJ49 are cold leg isolation valves and are directed to be closed in EOP-LOCA-6. Incorrect in that the SJ49s are used to isolate a leak on the RHR cold legs.

C. Incorrect. Plausible because 21 and 22 SJ49 are cold leg isolation valves and are directed to be closed in EOP-LOCA-6. Incorrect in that the SJ49s are used to isolate a leak on the RHR cold legs. Also the operator may believe that when the SI discharge pressure reads about the same RCS pressure that a leak exists.

D. Incorrect. Plausible because the 2SJ135 is closed per EOP-LOCA-6. Incorrect in that SI Pump discharge pressure alone does not necessarily indicate a leak, SI Pump flow is the best indicator because the RCS pressure is above the shutoff head of the SI pumps.

Technical References:	2-EOP-LOCA-6 (R41) & bases document
Proposed References to be provided:	None
Learning Objective:	NOS05LOCA06-05, Objective 5. Determine the indications that are monitored to ensure proper system/component operation for each step in the EOP for LOCA OUTSIDE CONTAINMENT
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.7 / 45.5 to 45.8
Level:	RO
Category/System:	10-LOCA Outside Containment
K/A #:	W E04 EA1.04 Ability to operate and/or monitor the following as they apply to LOCA Outside Containment: RCS
Importance:	3.8
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the operator how to determine a leak on the SI Pump Cold Leg based on RCS pressure and what valve is closed per EOP-LOCA-6 to isolate the leak on the SI Pump Cold Leg location.

17 **10-155** **Points: 1.00**

Given:

- Salem Unit 2 experienced a LOCA
- ALL RCPs have been stopped

Then:

- Loss of Emergency Recirculation exist
- The crew is implementing 2-EOP-LOCA-5, Loss of Emergency Coolant Recirculation
- RCS depressurization is in progress

Per 2-EOP-LOCA-5, which of the following completes both statements?

1. During the RCS depressurization upper head voiding __(1)__ expected.

2. Which of the following indications is monitored if head voiding occurs?

A. (1) is
(2) Core Exit Thermocouples

B. (1) is
(2) PZR level

C. (1) is NOT
(2) PZR level

D. (1) is NOT
(2) Core Exit Thermocouples

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because CET temperatures may increase due to head voiding. Incorrect in that EOP-LOCA-5 step 23 Note states that rapidly rising PZR level is a result of upper head voiding.

B. Correct. Per 2-EOP-LOCA-5 step 23, during RCS depressurization upper head voiding can be expected. As such, the operators need to monitor a rise in PZR level in order to stop the depressurization prior to the PZR going solid.

C. Incorrect. Plausible because the operator may believe that as long as the cooldown rate of 100 °F per hour is not exceeded then no possibility of upper head voiding will be expected.

D. Incorrect. Plausible because the operator may believe that as long as the cooldown rate of 100 °F per hour is not exceeded then no possibility of upper head voiding will be expected. Plausible because CET temperatures may increase due to head voiding. Incorrect in that EOP-LOCA-5 step 23 Note states that rapidly rising PZR level is a result of upper head voiding.

Technical References:	2-EOP-LOCA-5 (R43)
Proposed References to be provided:	None
Learning Objective:	NOS05LOCA05-06, Objective 6. Given any step, caution, note, or Continuous Action Summary for LOSS OF EMERGENCY COOLANT RECIRCULATION, describe the basis.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.7 / 45.7 / 45.8
Level:	RO
Category/System:	10- Loss of Emergency Coolant Recirculation
K/A #:	W E11 EK1.07 Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Loss of Emergency Coolant Recirculation: Upper head voiding during RCS depressurization
Importance:	3.7
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the applicant is being asked if upper head voiding is expected during the RCS depressurization and what parameter will indicate the presence of voiding.

Additional Reference:

EOP Step No: Note 23-1

ERG Step No: Note 23-1

EOP Step:

NOTE: THE UPPER HEAD REGION MAY VOID DURING RCS DEPRESSURIZATION IF RCPS ARE NOT RUNNING. THIS WILL RESULT IN A RAPIDLY RISING PZR LEVEL
[RCS DEPRESSURIZATION TO MINIMIZE SUBCOOLING]

Purpose:

To inform the operator that PZR level may increase rapidly during RCS depressurization with no RCPs running due to voiding in the upper head region.

ERG Basis:

Without RCPs running, there is very little flow into the upper head region. Liquid in that region remains relatively hot even though the liquid temperature in the active regions of the RCS has been significantly reduced during the RCS cooldown. As the RCS is subsequently depressurized, the hotter liquid in the upper head may flash to steam, forming a upper head void. Steam formation in the upper head will displace water into the PZR, causing rapidly increasing PZR level with the potential for water relief through the PZR PORVs. The PZR may fill with water within a few minutes. This note informs the operator of the potential for this condition, so that RCS depressurization can be stopped quickly to avoid a water solid PZR.

18 **10-156** **Points: 1.00**

Given:

- Salem Unit 2 has lost ALL feedwater
- The crew is performing 2-EOP-FRHS-1, Response to Loss of Secondary Heat Sink
- Safety Injection (SI) has been manually actuated and reset
- 24 SG is selected for depressurization
- The RO actuates MSLI on all loops except for 24
- 22MS167, Main Steam Isolation Valve, failed to close
- The PO adjusts the steam pressure valve demand to commence the depressurization

Based on the above conditions and in accordance with 2-EOP-FRHS-1, which of the following is the expected response and any required action if the depressurization rate is set too HIGH?

A. No effect on the depressurization since the Main Steam Line Isolation (MSLI) signals are blocked when SI was manually actuated and reset.

B. A High Steam Line Differential Pressure will result in a MSLI signal and isolate steam dumps; the crew will use atmospheric dumps to continue the depressurization.

C. A High Steam Flow with Low-Low-Tavg will result in a MSLI signal and isolate steam dumps; the crew will use atmospheric dumps to continue the depressurization.

D. The Low-Low-Tavg (P-12) signal will close all steam dumps; the crew will depress the bypass Tavg pushbuttons to continue the depressurization using ALL steam dump groups.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because the operator may incorrectly believe that manually actuating and resetting SI will block ALL MSLI signals (SI and MSLI). Incorrect in that the MSLI signal for high steam flow with low-low Tavg and hi-hi containment pressure can NOT be blocked..

B. Incorrect. Plausible because the high steamline differential pressure SI can NOT be blocked. Incorrect in that this signal does not actuate a MSLI.

C. Correct. A 100% steam dump valve dump will cause high steam flows and low-low Tavg temperature (P-12). P-12 will close the steam dumps, however, the high steam flow rate with P-12 will also actuate a MSLI closing all MSIVs resulting in no available steam dumps. The crew can continue the depressurization via the MS10s (atmospheric dumps).

D. Incorrect. Plausible because the operator may incorrectly believe that when SI was manually actuated and then reset, that MSLI signals are blocked. Incorrect in that MSLI signals can NOT be blocked.

Technical References:	2-EOP-FRHS-1 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05FRHS00-05, Objective 6.a Describe the plant response to actions taken in the following EOP's. 2-EOP-FRHS-1
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.7. / 45.7 / 45.8
Level:	RO
Category/System:	10-Loss of Secondary Heat Sink
K/A #:	W E05 EK1.09 Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to Loss of Secondary Heat Sink: MSLI on high steam pressure rate during cooldown.
Importance:	3.6
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the operator knowledge of MSLI signals and the cause and effect when operators are adjusting steam dumps to establish maximum rate of depressurization which can result in an unwanted MSLI and stopping the depressurization.

19 **11-210** **Points: 1.00**

Given:

- Salem Unit 2 has experienced a reactor trip
- The crew has transitioned to 2-EOP-TRIP-2, Reactor Trip Response
- RO reports three (3) control rods are NOT FULLY inserted
- The crew initiates a Rapid Boration via a BAT Pump
- 2CV175, Rapid Boration Valve, is fully OPEN
- Rapid Boration flowmeter indicates 20 gpm

Which of the following describes the NEXT required action per 2-EOP-TRIP-2?

A. Continue with rapid boration until the required amount of boron is achieved.

B. Stop the BAT pump, realign charging pump suction to the RWST, and maintain charging flow > 87 gpm.

C. Establish rapid boration via cold leg injection by realigning charging pump suction to the RWST and opening the BIT valves.

D. Direct equipment operator to locally open the 2CV174 to bypass the boric acid blender, fully open 2CV172 for maximum boron flow, and maintain charging flow > 87 gpm.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that adequate rapid boration flow is met. Incorrect in that rapid boron flow should be 36 gpm.

B. Correct. Per 2-EOP-TRIP-2 step 4, rapid boration is initially started using a BAT pump and then opening the 2CV175. Adequate rapid boration flow through the 2CV175 is 36 gpm. Based on the indicated boron flow, rapid boration flow is inadequate. As a result EOP-TRIP-2 will stop the BAT pump, realign the charging suction from the RWST.

C. Incorrect. Plausible because this action is performed in 2-EOP-FRSM-1 if inadequate rapid boration flow. Incorrect in that EOP-TRIP-2 does not direct this pathway.

D. Incorrect. Plausible because this action is performed for rapid boration when emergency borating from the field during control room evacuation.

Technical References:	2-EOP-TRIP-2 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05TRP002-08, Objective 5. Determine the indications that are monitored to ensure proper system/component operation for each step in EOP-TRIP-2
Question Source:	Bank – Byron 2019 NRC RO21
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	RO
Category/System:	11-Emergency Boration
K/A #:	APE 24 AA2.12 Ability to determine and/or interpret the following as they apply to Emergency Boration: Emergency boron flow
Importance:	3.5
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the question is asking to determine rapid boration flow is adequate and, if not, what is the next required action IAW EOP-TRIP-2.

20 **10-157** **Points: 1.00**

Given:

- Salem Unit 1 is at 100% power
- ALL Circulators are in service
- The BOP operator notices condenser backpressure is rising, but is unsure if it's valid

What other indication in the control room would confirm the rising backpressure and what is the concern if the conditions above continue?

A. Rising condenser differential temperature; damage to the Low Pressure turbine.

B. Rising condenser differential temperature; cavitation of the condensate pumps.

C. Lowering generator megawatts; cavitation of the condensate pumps.

D. Lowering generator megawatts; damage to the Low Pressure turbine.

Answer: D

Explanation / Justification:

A. Incorrect. First part is incorrect. Plausible because a rise in condenser differential temperature could indicate reduced heat transfer in the condenser resulting in a change in backpressure. Incorrect in that the best indication to look at is lowering MWs. Second part is correct.

B. Incorrect. First part is incorrect. Plausible because a rise in condenser differential temperature could indicate reduced heat transfer in the condenser resulting in a change in backpressure. Incorrect in that the best indication to look at is lowering MWs. Second part is incorrect. Plausible because with a rise in condenser differential temperature could affect condensate pump suction temperature. Incorrect in that overpressurization and damage to the condenser is the primary concern.

C. Incorrect. First part is correct. Second part is incorrect. Plausible because with a rise in condenser differential temperature could affect condensate pump suction temperature. Incorrect in that overpressurization and damage to the condenser is the primary concern.

D. Correct. Lowering condenser backpressure will cause the LP turbines to be less efficient and therefore result in a reduction or lowering of the main generator MWe output. The operational concern with a high backpressure is overpressurizing and damage to the main condenser.

Technical References:	NOS05MNTURB-14
Proposed References to be provided:	None
Learning Objective:	NOS05MNTURB-14, Objective 13.c. Given plant conditions, relate the Main Turbine System with the following: Main Condenser
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.5 / 41.7 / 45.7 / 45.8
Level:	RO
Category/System:	11-Loss of Condenser Vacuum
K/A #:	APE 51 AK1.02 Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to Loss of Condenser Vacuum: Relationship of condenser and M/TG operation
Importance:	3.7
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the candidate is being asked how condenser backpressure effects the output of the main turbine/generator and in addition what is the operational concern with high backpressure.

21 **10-158** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- Containment Pressure Channel III is inoperable and out of service for troubleshooting, The Channel has been properly removed from service IAW S2.OP-SO.RPS-0005, Placing Containment Pressure Channel in Tripped Condition

Then:

- Salem Unit 2 reactor tripped and SI actuated due to a LOCA
- The crew is implementing 2-EOP-TRIP-1, Reactor Trip or Safety Injection

The following conditions are now present:

- Containment pressure readings are:
  - Channel I – 15.2 psig
  - Channel II – 14.7 psig
  - Channel III – 0.0 psig
  - Channel IV – 14.5 psig

Based on the above conditions, what is the status of the Containment Spray system?

1. Conditions for Containment Spray and Phase 'B' Actuation __(1)__ met.
2. The Containment Spray Pump discharge valves (CS2s) will open automatically on a __(2)__ signal.

A. (1) are NOT
(2) Containment Spray

B. (1) are NOT
(2) Phase B

C. (1) are
(2) Phase B

D. (1) are
(2) Containment Spray

Answer: A

Explanation / Justification:

A. Correct. Conditions for Containment Spray and Phase B are not met based on 1/3 containment pressure channels > 15 psig. In this case with Channel III removed from service Channel III is bypassed which makes the logic 2/3 containment pressure channels are above 15 psig. Only Channel I is above the 15 psig therefore, no CS signal. The CS pump discharge valves (CS2s) open automatically on a CS actuation signal only.

B. Incorrect. Plausible because Phase B signal is actuated at the same 2/4 containment pressure logic. Incorrect in that only the CS signal actuates the CS valves.

C. Incorrect. Plausible because the operator may incorrectly recall the logic coincidence for CS spray and believe that you need 1/3 containment pressure signals present to actuate CS and Phase B based on Channel III removed from service. Plausible because Phase B signal is actuated at the same 2/4 containment pressure logic. Incorrect in that only the CS signal actuates the CS valves and when a containment pressure channel is removed from service, the channel is bypassed so that the logic goes from 2/4 to 2/3.

D. Incorrect. Plausible because the operator may incorrectly recall the logic coincidence for CS spray and believe that you need 1/3 containment pressure signals present to actuate CS and Phase B.

Technical References:	2-EOP-TRIP-1 (R42), NOS05CSPRAY-07
Proposed References to be provided:	None
Learning Objective:	NOS05CSPRAY-07, Objective 15.a. Given plant conditions relate the Containment Spray System with the following: RPS
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.7 / 45.7 / 45.8
Level:	RO
Category/System:	10-High Containment Pressure
K/A #:	W E14 EA1.08 Ability to operate and/or monitor the following as they apply to High Containment Pressure: CCS
Importance:	4.0
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the question is asking the applicant to monitor the containment pressure channels and determine if CS and Phase B have actuated and then determine which of the signals; CS or Phase B will open the CS pump discharge valves (CS2s).

22 **11-211** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- Indications of High RCS Activity exists
- The crew is implementing S2.OP-AB.RC-0002, High Activity in the RCS
- Chemistry sample of the RCS confirms the elevated RCS activity
- The RCS activity is <u>below</u> the TS limit

Which of the following describes the NEXT required action and the reason for the action?

A. Reduce letdown flow to minimize personnel radiation exposure in the Auxiliary Building.

B. Perform hourly RCS sampling to determine if TS limits are exceeded and a shutdown is required.

C. Maximize letdown flow to accelerate RCS activity cleanup through the demineralizers.

D. Shutdown the Unit within 6 hours to minimize the potential to release high radioactive airborne during a SG tube rupture.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because reducing letdown flow could reduce the dose rates in the Aux Building. Incorrect in that S2.OP-AB.RC-0002 does not direct reducing letdown flow, just the opposite letdown flow is increased.

B. Incorrect. Plausible because step 3.20 of S2.OP-AB.RC-0002 directs hourly sampling. Incorrect in that maximum letdown would be the next action by procedure.

C. Correct. Per S2.OP-AB.RC-0002 following confirmation of the RCS elevated activity and being below TS limits, the crew will maximize letdown flow to accelerate RCS cleanup through the demineralizers.

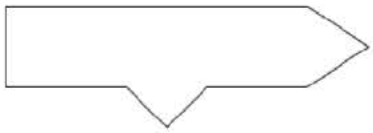
D. Incorrect. Plausible because this is the bases for TS 3.4.9, Specific Activity, to limit release during SGTR with primary-to-secondary leakage.

Technical References:	S2.OP-AB.RC-0002 (R12)
Proposed References to be provided:	None
Learning Objective:	NOS05ABRC02-07, Objective 2. Describe, in general terms, the actions taken in S2.OP-AB.RC-0002 and the bases for the actions in accordance with the Technical Bases Document.
Question Source:	Bank - Salem
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.10 / 45.6 / 45.13
Level:	RO
Category/System:	11-High Reactor Coolant Activity
K/A #:	APE 76 AK3.04 Knowledge of the reason for the following response and/or actions as they apply to High Reactor Coolant Activity: Maximizing demineralizer flow rates
Importance:	3.3
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the candidate is being asked the reason for maximizing letdown flow rates through the demineralizers during elevated RCS activity.

23 **10-160** **Points: 1.00**

When proceeding through 2-EOP-TRIP-3, SI Termination, you come across the following symbol at Step 1.



What does the symbol denote?

A. The stated action must be completed before continuing with the present flow path.

B. Initiate the stated action while continuing with the existing flow path.

C. The stated action shall be executed if or when the conditions specified occurs.

D. The crew must wait for the conditions specified to become established before proceeding on.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because this describes a "Sequential Branching" symbol which are used in EOPs.

B. Correct. Per OP-SA-108-101-2000 Figure 2, Flowchart Symbols, this symbol indicates a Concurrent Step symbol which requires the operator to enter the designated procedure and perform the stated actions while continuing in the existing flow path.

C. Incorrect. Plausible because this describes a "Conditional Step" symbol which are used in EOPs.

D. Incorrect. Plausible because this describes a "Wait Step" symbol which are used in EOPs.

Technical References:	OP-SA-108-101-2000 (R12)
Proposed References to be provided:	**Embedded symbol from EOP-TRIP-3**
Learning Objective:	NOS05CONDOP-15, Objective 5. Describe requirements for the following Control Room or Field Activities in accordance with applicable Conduct of Operations Manual Administrative Procedures.
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 45.13
Level:	RO
Category/System:	10-Rediagnosis & SI Termination / Emergency Procedures/Plan
K/A #:	W E01 & E02 G2.4.19 Knowledge of emergency and abnormal operating procedures layout, symbols, and icons.
Importance:	3.4
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the candidate is being asked knowledge of symbols used on EOPs and how the crew should implement them. This K/A asks about EOP-TRIP-7 (Re-diagnosis) and EOP-TRIP-3 (SI Termination), these are two different procedures and the question chose to use the symbol in EOP-TRIP-3 as being more discriminating than those symbols used in EOP-TRIP-7.

24 **10-161** **Points: 1.00**

Given:

- Salem Unit 1 experienced a reactor trip and automatic actuation of SI due to a LOCA.
- Containment pressure peaked at 20 psig and is now 13 psig and lowering
- The crew is continuing with 1-EOP-LOCA-1, Loss of Reactor or Secondary Coolant, following the successful transfer to Cold Leg recirculation
- Due to containment high water level, the crew has entered 1-EOP-FRCE-2, Response to Containment Flooding

Which of the following systems is the cause of the unexpected water level in containment?

A. Fire Protection

B. Service Water

C. Component Cooling Water

D. Primary Water

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the Fire Protection system is a source of water into containment per 1-EOP-FRCE-2. Incorrect in that FP system is isolated on Phase A signal.

B. Correct. Service Water from the CFCUs does not get isolated by Phase A or B signals. Therefore, SW system is the only source of water into containment following a Phase A and B signals.

C. Incorrect. Plausible because the Component Cooling Water system is a source of water into containment per 1-EOP-FRCE-2. Incorrect in that CCW system is isolated on Phase B signal since containment pressure greater than 15 psig

D. Incorrect. Plausible because the Primary Water system is a source of water into containment per 1-EOP-FRCE-2. Incorrect in that PW system is isolated on Phase A signal.

Technical References:	1-EOP-FRCE-2 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05FRCE00-08, Objective 1.2. Describe the EOP mitigation strategy for the following: 1-EOP-FRCE-2
Question Source:	Bank – Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.8 / 41.10 / 45.3
Level:	RO
Category/System:	10-Containment Flooding
K/A #:	W E15 EK2.03 Knowledge of the relationship between Containment Flooding and the following systems or components: Service water
Importance:	3.3
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the candidate is being asked the sources of water into containment that could be causing flooding in containment. The applicant also needs to know which system is not isolated when a Phase A and B signals are actuated.

25 **10-162** **Points: 1.00**

Given:

- Salem Unit 2 experienced a reactor trip from 100% power

Then:

- STA reports a valid CFST YELLOW path for Containment Environment based on high radiation readings in containment
- The CRS enters 2-EOP-FRCE-3, Response to High Containment Radiation Levels

Which of the following completes the statement concerning the mitigating strategy that the 2-EOP-FRCE-3 will direct?

Operate ALL CFCUs in __(1)__ speed through the __(2)__ filters to reduce the radioactivity in the containment atmosphere.

A. (1) High
(2) Roughing

B. (1) High
(2) HEPA

C. (1) Low
(2) HEPA

D. (1) Low
(2) Roughing

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because the CFCUs can be operated in HIGH speed and the operator may believe that in HIGH speed higher air flow will pass thru the filters for better cleanup.

B. Incorrect. Plausible because the CFCUs can be operated in HIGH speed and the operator may incorrectly believe that the dampers to redirect the air flow thru the HEPA filters can be operated from the control room. Incorrect in that the dampers positions on the console is indication only to tell the operators if the HEPA filters is in service.

C. Correct. Per 2-EOP-FRCE-3 step 2, the crew is directed to place all CFCUs is LOW speed. In LOW speed operation the dampers will realign flow through the HEPA filters and bypass the roughing filters to reduce the radioactivity inside containment.

D. Incorrect. Plausible because LOW speed is the required operation of the CFCUs. The operator may confuse which type of filter is used. During LOW and HIGH speed operation.

Technical References:	2-EOP-FRCE03 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05FRCE00-08, Objective 3. Identify possible radioactivity release paths for a containment high pressure or high radiation condition, and describe how the actions in 2-EOP-FRCE-1 and 3 minimize the release potential.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.8 / 45.5 to 45.8
Level:	RO
Category/System:	11-High Containment Radiation
K/A #:	W E16 EA1.05 Ability to operate and/or monitor the following as they apply to High Containment Radiation: Containment atmosphere filtration system
Importance:	3.2
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the question is asking the applicant how will the CFCUs be operated in order to place the atmosphere filtration system in service (HEPA).

26 **11-212** **Points: 1.00**

Given:

- A Loss of All Off-site power has occurred at Salem Unit 2.
- The crew is performing 2-EOP-TRIP-4, Natural Circulation.

Then:

- Power has been restored to all the 4KV Group Buses.

Per 2-EOP-TRIP-4, in the order of priority how many RCP(s) is/are started and what is the primary reason for starting the RCP(s) during natural circulation?

A. One; Prevents two phase flow from occurring and impeding natural circulation.

B. Two; Prevents two phase flow from occurring and impeding natural circulation.

C. Two; Permits a faster plant cooldown with less potential for upper head voiding.

D. One; Permits a faster plant cooldown with less potential for upper head voiding.

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because this would be the reason for stopping an RCP during small break LOCAs. Incorrect in that this is not the primary reason stated in EOP-TRIP-4 bases document.

B. Incorrect. Plausible because two RCPs could be started for better spray if 23 RCP is not available. Incorrect in that EOP-TRIP-4 Note N1-1 does allow two RCPs to start to support better spray if 23 RCP is not available. Plausible because this would be the reason for stopping an RCP during small break LOCAs. Incorrect in that this is not the primary reason stated in EOP-TRIP-4 bases document.

C. Incorrect. Plausible because two RCPs could be started for better spray if 23 RCP is not available. Incorrect in that EOP-TRIP-4 Note N1-1 does allow two RCPs to start to support better spray if 23 RCP is not available.

D. Correct. Per 2-EOP-TRIP-4 Bases Document Note (N1-3), RCPs would be started whenever possible during the course of EOP-TRIP-4. Forced convection cooling permits a faster plant cooldown with less potential for upper head voiding, an attempt to restart RCPs should be made when under natural circulation conditions.

Technical References:	2-EOP-TRIP-4 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05TRP004-08, Objective 4. Determine the basis for each step, caution, note or continuous action step relative to a NATURAL CIRCULATION COOLDOWN
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.5 / 41.10 / 45.6 / 45.13
Level:	RO
Category/System:	10-Natural Circulation
K/A #:	W E09 EK3.06 Knowledge of the reasons for the following responses and/or actions as they apply to Natural Circulation Operation: Starting an RCP
Importance:	3.4
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the candidate is being asked how the reason for starting an RCP during natural circulation conditions per EOP-TRIP-4.

Additional Reference:

EOP Step No: Note 1-1

ERG Step No: Note 1-2

EOP Step:

NOTE: RCPs SHOULD BE RUN IN ORDER OF PRIORITY TO PROVIDE NORMAL PZR SPRAY (23, 21+22 OR 21+24, 21)
[RCP RESTART]

Purpose:

To inform the operator of a preferred order for starting RCPs.

ERG Basis:

For the reference plant there are PZR connections to one RCS hot leg via the surge line and to two RCS cold legs via the spray lines. Single pump operation in the loop that provides the best spray is preferred to obtain normal PZR spray capability. If the RCP in the loop with the PZR surge line can be started, then it alone would be sufficient to provide normal PZR spray. However, if that RCP is unavailable, it will likely be necessary to start more than one RCP to provide normal PZR spray. Refer to the document RCP TRIP/RESTART in the Generic Issues section of the ERG Executive Volume.

Additional Reference:

EOP Step No: Note 1-2

ERG Step No: Note 1-3

EOP Step:

NOTE: IF CONDITIONS CAN BE ESTABLISHED FOR STARTING AN RCP DURING THIS PROCEDURE, STEP 1 SHOULD BE REPEATED
[RCP RESTART]

Purpose:

To inform the operator that an RCP should be started whenever possible during the course of this procedure, and the guidance in Step 1 should be used.

ERG Basis:

Since forced convection cooling permits a faster plant cooldown with less potential for upper head voiding, an attempt to restart an RCP should be made when under natural circulation conditions. If the proper conditions can be established for starting an RCP, Step 1 should be repeated. Step 1 provides conditions necessary for starting an RCP and should be used when attempting a restart. This step also directs the operator to the appropriate procedure if restart is successful.

27 **11-213** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

Then:

- OHA D-20, 21 RCP BRG CLG WTR FLO LO, is Alarming
- RO reports that all RCP motor bearing temperatures are rising

Which of the following completes the statement?

Per S2.OP-AB.RCP-0001, RCP Abnormality, the crew has a MAXIMUM of __(1)__ minutes to restore CCW flow OR is required to trip the reactor and stop the RCP anytime RCP motor bearing temperatures exceeds __(2)__.

A. (1) 2
(2) 175 °F

B. (1) 2
(2) 165 °F

C. (1) 5
(2) 165 °F

D. (1) 5
(2) 175 °F

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may confuse the 2 minutes requirement with the loss of seal injection flow and loss of CCW flow to the thermal barrier.

B. Incorrect. Plausible because 165 °F is the limit in the ARP for OHA D-20 that directs transition to S2.OP-AB.RCP-0001. Also the operator may confuse the 2 minutes requirement with the loss of seal injection flow and loss of CCW flow to the thermal barrier.

C. Incorrect. Plausible because 165 °F is the limit in the ARP for OHA D-20 that directs transition to S2.OP-AB.RCP-0001.

D. Correct. Per S2.OP-AB.RCP-0001, Attachment 1, the crew has up to 5 minutes to restore the loss of CCW or a reactor trip is required and the RCP must be stopped. Also anytime if RCP motor bearing temperatures exceed 175 °F, then the reactor must be tripped and the affected RCP stopped.

Technical References:	S2.OP-AB.RCP-0001 (R28)
Proposed References to be provided:	None
Learning Objective:	NOS05RCPUMP-18, Objective 10.b, State the Technical Specification associated with the component, parameters and operation of the Reactor Coolant Pump including the Limiting Condition for Operation(s) (LCO) and the applicability of the LCO(s)
Question Source:	Bank – Beaver Valley 2018 NRC RO29
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.12 / 45.13
Level:	RO
Category/System:	11-Reactor Coolant Pump System / Conduct of Operations
K/A #:	003 G2.1.7 Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation.
Importance:	4.4
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked to evaluate performance data from the RCP seal injection surveillance test and then determine if the parameters recorded from the test meets the TS LCO.

Additional Reference:

s2.OP-AB.RCP-0001(Q)

ATTACHMENT 1
(Page 1 of 2)

CONTINUOUS ACTION SUMMARY

1.0 IF AT ANY TIME, any of the following validated RCP conditions exist, THEN GO TO Attachment 2, Stopping Reactor Coolant Pumps.

_____
Time

- Component Cooling Water flow is lost to ALL RCPs.
- Component Cooling Water flow not restored within 5 minutes of initial loss:
  - OHA-D-20, 21 RCP BRG CLG WTR FLO LO
  - OHA-D-21, 22 RCP BRG CLG WTR FLO LO
  - OHA-D-22, 23 RCP BRG CLG WTR FLO LO
  - OHA-D-23, 24 RCP BRG CLG WTR FLO LO
- RCP Seal Injection Flow AND RCP Thermal Barrier Component Cooling flows are lost concurrently (RCPs should be secured within 2 minutes to prevent RCP damage).
- Shaft vibration greater than 20 mils or motor flange vibration greater than 5 mils **[C0379]**
- Motor Bearing temperature greater than 175°F
- Motor Winding temperature greater than 302°F, (Modes 1-5)
  (Also, REFER TO Attachment 3 (page 2 of 2) for Alternate Indications to validate indicated rise in RCP Motor Winding Temperature)
- Seal Water Outlet temperature greater than 190°F
- #1 Seal Water Leakoff flow less than 0.8 gpm and reverses to greater than or equal to 6.0 gpm or pump bearing/seal inlet temperature rise
- #1 Seal Water Leakoff flow less than 0.8 gpm with rising pump bearing/seal inlet temperatures
- #1 Seal Water Leakoff flow greater than or equal to 6 gpm

28 **01-017** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- 2CC71, Letdown HX Temp Control Valve, is in Auto

Then:

- 2TE-130, temperature detector for 2CC71, fails LOW

Which of the following describes the impact of this failure?

1. Letdown temperature will __(1)__ and the RCS T-avg temperature will __(2)__.

A. (1) rise (2) lower

B. (1) rise (2) rise

C. (1) lower (2) rise

D. (1) lower (2) lower

Answer: A

Explanation / Justification:

A. Correct. When the 2TE-130 temp detector to the 2CC71 fails low, it will cause the 2CC71 to go closed and the Letdown HX outlet temperature will rise, the hotter letdown flowing through the Mixed Bed Demineralizers will cause boron to be released into the RCS resulting in a boration effect and lowering T-avg temperature.

B. Incorrect. Plausible because the operator may incorrectly recall how temperature affects the boron affinity of a mixed bed demineralizer and believe a higher temp absorbs boron.

C. Incorrect. Plausible because the operator may confuse the operation of the 2CC71 when the temperature detector fails low and believe that the 2CC71 will open. Plausible because the operator may incorrectly recall how temperature affects the boron affinity of a mixed bed demineralizer and believe a higher temp absorbs boron

D. Incorrect. Plausible because the operator may confuse the operation of the 2CC71 when the temperature detector fails low and believe that the 2CC71 will open.

Technical References:	INPO 191007 Demineralizers
Proposed References to be provided:	None
Learning Objective:	NOS05DEMINE-02, INPO 191007
Question Source:	Bank – Salem
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 45.5
Level:	RO
Category/System:	01-CVCS
K/A #:	004 A3.06 Ability to monitor automatic features of the CVCS system, including: T-ave and T-ref
Importance:	3.6
Tier/Group	Plant Systems - Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked how the Auto function of the 2CC71 is affected when the temp detector fails low and how T-ave is affected.

29 **02-031** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- 21 Charging Pump is in service
- 2LT-112, VCT level, transmitter fails HIGH

Which of the following describes the operational impact to the CVCS system and the effect on the running charging pump, if any (assume no operator action taken)?

A. No Auto make-up, VCT level lowers, Auto swap to RWST; no effect on 21 charging pump.

B. No Auto make-up, VCT level lowers, no Auto swap to RWST; subsequent loss of suction and damage to 21 charging pump.

C. Auto make-up maintains VCT level above the low-low level swapover; no effect on 21 charging pump.

D. Auto make-up initiates but unable to maintain level in VCT, no Auto swap to RWST; subsequent loss of suction and damage to 21 charging pump.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible if the operator incorrectly recalls that auto swapover will actuate when 1 out of 2 VCT levels are low-low. Incorrect in that auto swapover requires 2 out of 2 VCT levels low-low..

B. Correct. 2LT-112 failing high will cause the CV35 high level divert valve to open. This will result in actual VCT level lowering. With the LT-112 failed high, auto make-up will not actuate and auto swapover to the RWST on VCT low-low level will not function due to needing 2 out of 2 VCT levels low-low to initiate auto swapover. As VCT continues to lower, the inservice centrifugal charging pump will lose suction and eventually run drying causing pump damage.

C. Incorrect. Plausible if the operator believes that 2LT-114 will also actuate auto makeup. Incorrect in that only 2LT-112 will actuate auto make-up when VCT low level is reached.

D. Incorrect. Plausible if the operator believes that 2LT-114 will also actuate auto make-up and that 1 out 2 VCT levels will cause auto swapover. Incorrect in that only 2LT-112 actuates auto make-up and auto swapover requires 2 out of 2 VCT levels low-low.

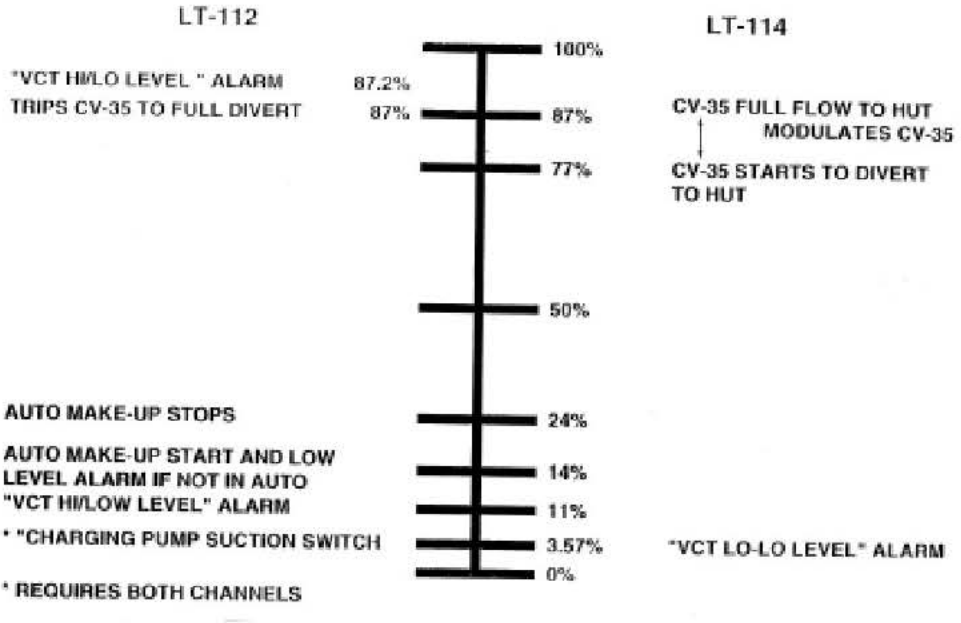
Technical References:	NOS05CVCS00-18
Proposed References to be provided:	None
Learning Objective:	NOS05CVCS00-18, Objective 15. Given plant conditions, relate the Chemical and Volume Control System and the following: Emergency Core Cooling System
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 / 45.4
Level:	RO
Category/System:	02-CVCS
K/A #:	004 K3.13 Knowledge of the effect that a loss or malfunction of the CVCS system will have on the following systems or system parameters: ECCS.
Importance:	3.8
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked how a failure of a VCT level transmitter will affect the in-service charging pump.

Additional Reference:

S2.OP-AB.CVC-0001(Q)

EXHIBIT 2
VCT LEVEL CONTROLLER FUNCTIONS



30 **04-072** **Points: 1.00**

Given:

- Salem Unit 1 is in MODE 4 performing a shutdown.
- 11 RHR Pump is in service
- 11 RHR Heat Exchanger is in service
- 11RH18 and 12RH18, RHR HX Flow Cont. Valves, are being throttled
- 1RH20, RHR HX Bypass Valve, is being throttled
- The BOP operator has determined that the cooldown is approaching the maximum allowable Technical Specification limit

Which of the following describes the actions that the operator will take to LOWER the cooldown rate while maintaining a constant RHR flow rate?

A. Lower the demand on the RH18s and Raise the demand on the 1RH20.

B. Raise the demand on the RH18s and Lower the demand on the 1RH20.

C. Lower the demand on the RH18s and Lower the demand on the 1RH20.

D. Raise the demand on the RH18s and Raise the demand on the 1RH20.

Answer: A

Explanation / Justification:

A. Correct. In order to lower the cooldown rate and maintain current flowrate thru the 11 RHR HX, the operator will need to lower the demands on the RH18s to close them more to reduce flow thru the 11 RHR HX and raise the demand on 1RH20 to raise the bypass flow around the 11 RHR HX.

B. Incorrect. Plausible if the operator believes that raising demand on the RH18s will close the valves and lowering demand on the RH20 will open the valve. Incorrect in that raising demand on the RH18s opens them more and lowering demand on the RH20 will close the valve resulting in raising the cooldown rate.

C. Incorrect. Plausible because lowering the demands on the RH18s and the RH20 will lower the cooldown rate but will also lower the overall RHR flowrate.

D. Incorrect. Plausible if the operator believes raising the demands on the RH18s and the RH20 will lower the cooldown rate. Incorrect in that raising the demands will open the RH18s more and the RH20 more resulting in rise in cooldown or possible not effect on the cooldown rate.

Technical References:	NOS05RHR000-19
Proposed References to be provided:	None
Learning Objective:	NOS05RHR000-19, Objective 4. Describe how the following components impact the Residual Heat Removal System during normal and abnormal conditions: RH18s and RH20
Question Source:	Modified – Salem 2020 NRC RO31
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 / 45.5 to 45.8
Level:	RO
Category/System:	04P-RHR System
K/A #:	005 A4.02 Ability to manually operate and/or monitor the RHR System in the control room: RHR heat exchanger temperature/bypass control valves
Importance:	3.9
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked how to manually operate the RH18s and RH20 (RHR HX bypass valve) to control the cooldown rate.

Parent Question: Salem 2020 NRC RO31

Modified question based on the stem asking about the cooldown.

Exam Outline Cross Reference:	Level:	RO SRO
	Tier #	2
	Group #	1
	K/A #	005000K6.03 Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: RHR heat exchanger.
Importance:		2.5

Question: #31

Given:

- Unit 2 is in MODE 5.
- 21 RHR HX Loop is providing Shutdown Cooling with a cooldown rate of 20 °F/hr.
- 21RH18, RHR HX Flow Control Valve, is throttled at 40 % to maintain total flow at 2500 gpm.
- 2RH20, RHR HX Bypass Valve, is throttled at 30 % open.

Subsequently the following occurs:

- Foreign material dropped in the Refueling Cavity is introduced into the RHR system causing partial blockage on the 21 RHR HX tubes.
- Cooldown rate has lowered to 5 °F/hr.

Based on the above conditions, in order to maintain a 20 °F/hr cooldown rate, the operator will be required to _____. (Assume decay heat load remains constant)

A. Lower the demand on the 21RH18 and Lower the demand on the 2RH20.

B. Lower the demand on the 21RH18 and Raise the demand on the 2RH20.

C. Raise the demand on the 21RH18 and Raise the demand on the 2RH20.

D. Raise the demand on the 21RH18 and Lower the demand on the 2RH20.

Answer: D

31 **02-032** **Points: 1.00**

Which of the following RCS break locations results in the highest peak clad temperature during a small break LOCA?

A. Cold Leg

B. Hot Leg

C. CRDM on Reactor Head

D. Pressurizer Steam Space

Answer: A

Explanation / Justification:

A. Correct. Due to thermodynamics, a cold leg break gives the highest peak clad temperatures. Also, ECCS pumps inject into the cold legs.

B. Incorrect. Plausible because a hot leg break will result in the highest loss of flow. Incorrect in that this accident does not result in the highest peak clad temperatures.

C. Incorrect. Plausible because a Control Rod Ejection is an analyzed SBLOCA. Incorrect in that this accident does not result in the highest peak clad temperatures.

D. Incorrect. Plausible because vapor space accidents at the PZR is an accident analyzed and located on the hot leg. Incorrect in that this accident does not result in the highest peak clad temperatures.

Technical References:	NOS05ECCS00-18
Proposed References to be provided:	None
Learning Objective:	NOS05ECCS00-18, Objective 1. Describe the purpose of the Emergency Core Cooling System.
Question Source:	Bank – Braidwood 2018 RO41
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.3
Level:	RO
Category/System:	02-Emergency Core Cooling System
K/A #:	006 K5.07 Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to ECCS: Expected temperature values in various locations of the RCS due to different break locations during all ECCS injection modes
Importance:	3.1
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked how what RCS break location represents the worst case scenario from an accident analysis standpoint.

32 **05-021** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- Console Alarm PRT LEVEL HI-LO is LIT
- PRT level is 90%

Which of the following completes the statement?

PRT level will be lowered…

A. via RCDT Pump to the Auxiliary Building Sump Tank

B. via RCDT Pump to a CVCS Hold-Up Tank

C. by gravity drain to a CVCS Hold-Up Tank

D. by gravity drain to the Auxiliary Building Sump Tank

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because most drain in the Aux Building drain to the Aux. Bldg. Sump Tank. Incorrect in that the PRT does not drain to the Aux. Bldg. Sump Tank.

B. Correct. To lower level in the PRT, S2.OP-SO.PZR-0003 will drain the PRT by opening the 2PR14 which will automatically open the 2WL12 which starts a RCDT Pump and defeats the low level cu-off. The RDCT pumps can discharge to either (1) RWST, (2), CVCS HUTs, or (3) WHUTs.

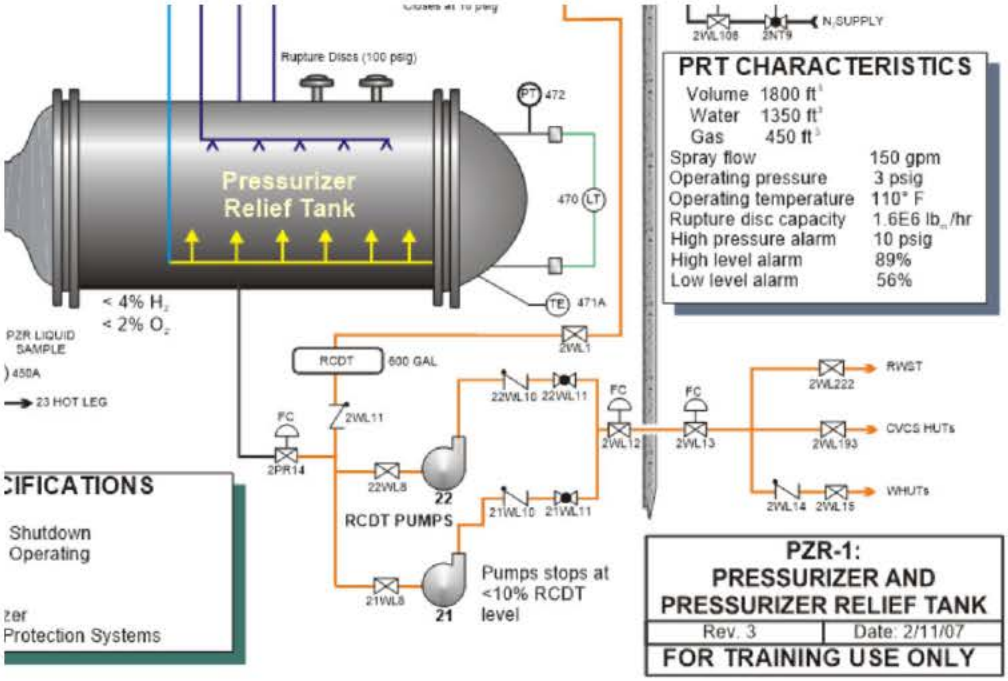
C. Incorrect. Plausible because the operator may believe that the RCDT Pumps are only used to drain down the RCDT and that gravity drain is the method used to drain down the PRT.

D. Incorrect. Plausible because the operator may believe that the RCDT Pumps are only used to drain down the RCDT and that gravity drain is the method used to drain down the PRT.

Technical References:	S2.OP-SO.PZR-0003 (R17), Simplified PZR-1
Proposed References to be provided:	None
Learning Objective:	NOS05PZRPRT-09, Objective 4.g. Describe the function of the following components and how their normal and abnormal operation affects the Pressurizer and Pressurizer Relief Tank: Pressurizer Relief Tank
Question Source:	New
Question Cognitive Level:	Fundamental/Analysis
10CFR Part 55 Content:	41.2 to 41.9 / 45.7 to 45.8
Level:	RO
Category/System:	05-Pressurizer Relief/Quench Tank System
K/A #:	007 K1.06 Knowledge of the physical connections and/or cause and effect relationships between the PRT System and the following systems: LRS
Importance:	2.6
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked how the PRT system is physically connected to the LRS. In this case, the PRT is drained to LRS via CVCS Hold-up Tanks or Waste Hold-up Tanks.

Additional Reference:



33 **08-021** **Points: 1.00**

Given:

- Salem Unit 2 experienced a LOCA and a Loss of Off-site Power
- The crew is implementing 2-EOP-TRIP-1, Reactor Trip or Safety Injection

Which of the following describes the operational impact on the Component Cooling Water System (CCW) and the reason why?

A. The CCW Pumps previously running remain running and both CCW heat exchangers are available; no likelihood for SW pump runout.

B. NO CCW Pumps are running and Service Water (SW) to the CCW heat exchangers is automatically isolated to prevent SW pump runout and diesel generator overloading.

C. NO CCW Pumps are running and SW to the CCW heat exchangers must be manually isolated to prevent SW pump runout and diesel generator overloading.

D. ALL CCW Pumps will be running and SW to the CCW heat exchangers is isolated to prevent excessive SW pump runout.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because this is how the CCW system operates when the SECs are operating in Mode 1 condition (SI only). Incorrect on that the SECs are in Mode 3 (SI plus blackout).

B. Correct. The plant is in SEC Mode 3 (SI plus Blackout) condition which will trip all CCW Pumps and isolate (close) service water flow to both CCW heat exchanger inlet valves. The SW is isolated to prevent runout conditions with only three SW pumps running and the CCW Pumps do not get sequenced onto the diesel due to loading concerns on the diesel generator. A CCW Pump is started per 2-EOP-APPX-1 to support safeguard loads and RHR HX operation during Cold Leg recirculation operation.

C. Incorrect. Plausible because no CCW pumps are running during a SEC Mode 3 condition and SW is required to be isolated to prevent SW pump runout. Incorrect in that the SW flow to the CCW HXs are automatically isolated during an SEC Mode 3 condition.

D. Incorrect. Plausible because this is how the CCW system operates when the SECs are operating in Mode 2 condition (Blackout only). Incorrect on that the SECs are in Mode 3 (SI plus blackout).

Technical References:	NOS05CCW000-15
Proposed References to be provided:	None
Learning Objective:	NOS05CCW000-15, Objective 16.e. Given plant conditions, relate the Component Cooling Water System with the following: Safeguards Equipment Controller (SEC)
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.3
Level:	RO
Category/System:	08-Component Cooling Water System
K/A #:	008 K5.10 Knowledge of the operational implications of the following concepts as they apply to the CCWS: Requirements on and for the CCWS for different conditions of the power plant
Importance:	3.1
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked the operational impact on the CCW Pumps during an Accident plus Blackout plant condition and the reason why.

34 **03-026** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- Master Pressure Controller (MPC) is in AUTO
- PZR Pressure Channel I is selected for Control
- PZR Pressure Channel IV is selected for Alarm

Then:

- A malfunction with the MPC causes the output demand to fail LOW (0%)

Which of the following identifies the effect on RCS pressure and the expected system response, assuming no operator action?

RCS pressure will…

A. lower and NO PZR Backup Heaters will energize.

B. lower and BOTH PZR Backup Heaters will energize.

C. rise and NO PZR PORVs will open.

D. rise and BOTH PZR PORVs will open.

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may incorrectly recall how the MPC demand effects the operation of the PZR heaters and spray valves. Incorrect in that a 0% demand from the MPC will energize both PZR heater backup groups and cause RCS to rise.

B. Incorrect. Plausible because the operator may incorrectly recall how the MPC demand effects the operation of the PZR heaters and spray valves. Incorrect in that a 0% demand from the MPC will energize both PZR heater backup groups and cause RCS to rise.

C. Incorrect. Plausible because the operator may incorrectly interpret how the MPC demand effects the operation of the PZR PORVs and believe with the demand failed low no PZR PORVs will open. The figure in S2.OP-AB.PZR-0001 Attachment 2 can mislead the operator thinking that the MPC does have some control function on the PZR PORVs. Incorrect in that the MPC does not control the PZR PORVs.

D. Correct. When the Master Pressure Controller (MPC) output fails low (0% demand), spray valves will close and both PZR Backup Heaters will energize raising RCS pressure. As RCS pressures rises, PZR sprays will not open since they are controlled from the MPC. Both PZR PORVs will open when RCS pressure exceeds 2335 psig.

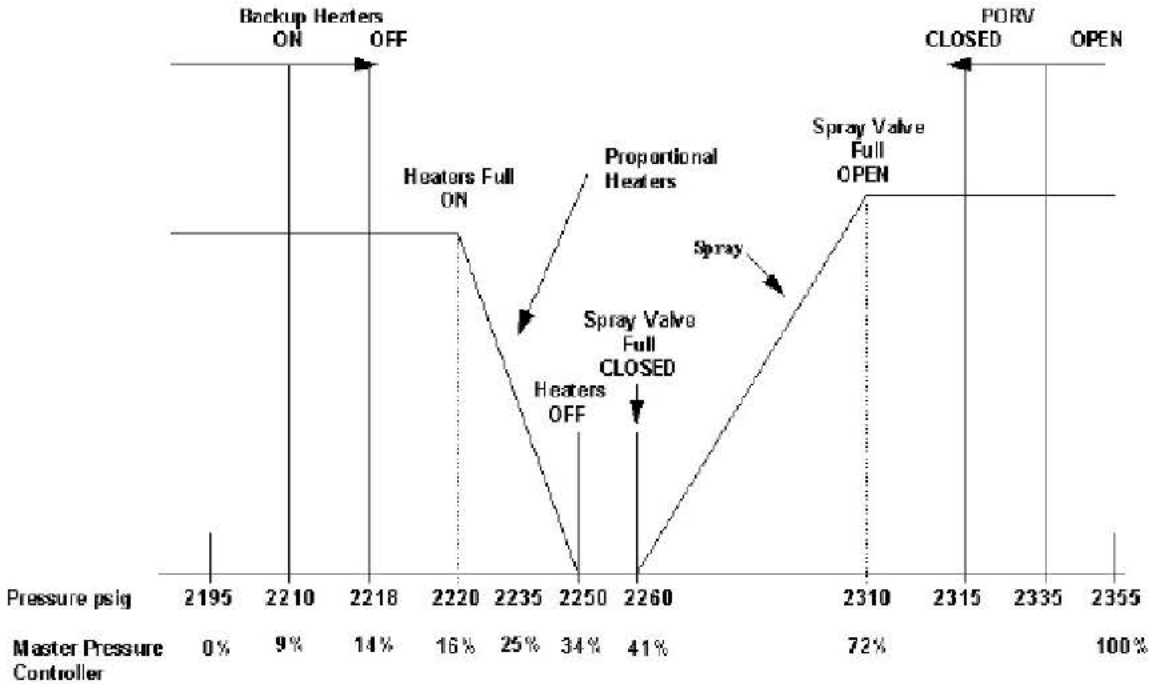
Technical References:	NOS05PZRPL-12, S2.OP-AB.PZR-0001 (R20)
Proposed References to be provided:	None
Learning Objective:	NOS05PZRPL-12, Objective 9. State the setpoints, coincidence, blocks and permissives for automatic actuations associated with the Pressurizer Pressure and Level Control System
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7
Level:	RO
Category/System:	03-Pressurizer Pressure Control System
K/A #:	010 K4.03 Knowledge of PZR Pressure Control System design features and/or interlocks that provide for the following: Overpressure control
Importance:	3.9
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked the design feature with the operation of the PZR PORVs for overpressure control. Specifically, the candidates is being tested on how a Master Pressure Controller fails low will effect RCS pressure and whether the PZR PORVs will open. Some plant designs operate the PORVs from the MPC controller. Salem's design the PZR PORVs are interlocked directly from the PZR pressure channels.

Additional Reference: S2.OP-AB.PZR-0001

ATTACHMENT 2
(Page 1 of 1)

PRESSURIZER PRESSURE CONTROL



35 **07-024** **Points: 1.00**

Given:

- Salem Unit 2 is at 7% power

Which of the following conditions would DIRECTLY result in a reactor trip?

A. Three FLOW LO bistables tripped on one RC Loop.

B. Three PZR PRESS LO bistables tripped.

C. Two PZR PRESS HI bistables tripped.

D. One SG LO LEVEL bistable tripped on two SGs.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because the reactor trip logic is met based on 2/3 flows < 90% on 1/4 RC Loops. Incorrect in that this reactor trip is enabled when reactor power is above P-8 (36%). Note, above P-7 requires 2/4 RC loop flows < 90%.

B. Incorrect. Plausible because the reactor trip logic is met based on 2/4 PZR pressure < 1865 psig. Incorrect in that this reactor trip is blocked until reactor power is above P-7 (10%).

C. Correct. PZR Pressure High on 2/4 channels > 2385 psig will cause a reactor trip. This trip is not blocked in any plant condition.

D. Incorrect. Plausible because the operator may confuse the trip logic as 1/3 NR level low on 1/4 SGs. Incorrect in that reactor trip requires 2/3 NR levels low on 1/4 SGs.

Technical References:	Fluency List
Proposed References to be provided:	None
Learning Objective:	NOS05RXPROT-15, Objective 12.a. State the setpoints, coincidence, blocks and permissives for all Reactor Trips and Safety Injections actuations
Question Source:	Bank – Point Beach 2017 RO1
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 / 45.7
Level:	RO
Category/System:	07-Reactor Protection System
K/A #:	012 A3.06 Ability to monitor automatic features of the RPS including: Trip logic
Importance:	4.1
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the applicant is being asked to monitor the given plant conditions to determine if an automatic reactor tripped has occurred.

Additional Reference:

VI. REACTOR TRIP SETPOINTS

SLIDE-17; ELO-12 (Operator Fundamental – Closely Monitor Plant Indications)

Name	Setpoint	Coincidence	Permissive
Manual	Manual	1/2 Switches	None
SI	Any SI	1 key switch ("S" signal) on either train	None
SR High Flux	1.0×10^5 cps	1/2 SR Detectors	P-6 allows block
IR High Flux	25 %	1/2 IR Detectors	P-10 allows block
PR High Flux LO SP	25%	2/4 PR Detectors	P-10 allows block
PR High Flux Hi SP	109%	2/4 PR Detectors	None
PR High Positive Rate	+5 % of RTP w/2 sec TC	2/4 PR Detectors	None
OTΔT	Variable[1]	2/4 ΔT's	None
OPΔT	Variable[2]	2/4 ΔT's	None
SG LO-LO Level	14 %	2/3 levels on 1/4 SG's	None
RCP Breaker Open	2 Breakers Open	2 RCP breakers	> P-7
Loss of Flow	<90%	2/3 on 1/4 RC Loops	> P-8
Loss of Flow	<90%	2/3 on 2/4 RC Loops	> P-7
RCP Undervoltage	<70% for >0.1 sec	1/2 on Group Buses H or E (AND) 1/2 on Group Buses F or G	> P-7
RCP Underfrequency	<56.5Hz, >0.1 sec	Same as above	> P-7
PZR Pressure High	>2385 psig	2/4 PZR Press Channels	None
PZR Pressure Low	<1865 psig	2/4 PZR Press Channels (rate compensated)	> P-7
PZR Level High	>92%	2/3 PZR Level Channels	> P-7
Turbine Trip	<45 psig ASOP[3] or 4/4 stop valves closed	2/3 channels for <45 psig or Stop valves 15% closed from open limit	> P-9

[1] $OT\Delta T_{setpoint} = \Delta T_0 \{ K1 - K_2 [(1 + \tau_1 s) \div (1 + \tau_2 s)] (T - T') + K_3 (P - P') - f_1(\Delta\phi)\}$

[2] $OP\ \Delta T_{setpoint} = \Delta T_0 \{K_4 - K_5 [(\tau_3 s \div (1 + \tau_3 s)] T - K_6 (T - T'') - f_2(\Delta\phi)\}$

36 **07-025** **Points: 1.00**

Given:

- Salem Unit 2 is at 18% Power
- 21 and 22 SPTs are supplying the 4KV Group Buses

Then:

- An electrical transient occurs on the 500 KV grid.
- During the transient the following values were reached:

4KV Group Bus	H	E	F	G
Frequency (Hz)	56.0	56.0	59.7	59.7
Control Room Indication (VAC)	4100	4100	4200	4200

Which ONE of the following describes the expected status of the reactor trip breakers and the Reactor Coolant Pumps (RCPs) assuming no operator action?

A. Reactor Trip breakers remain closed and ALL RCPs remain running.

B. Reactor Trip breakers remain closed and ONLY 2 RCPs remain running.

C. Reactor Trip breakers open and ALL 4 RCPs trip off.

D. Reactor Trip breakers open and ONLY 2 RCPs remain running.

Answer: A

Explanation / Justification:

A. Correct. The reactor trip logic from an RCP UF/UV is based on 1 out of 2 buses taken twice at 56.5 Hz (H bus or E bus AND F bus or G bus) trips all four RCPs and if above P-7 (10% power), initiates a reactor trip. In this case, the reactor trip logic is not satisfied since the underfrequency condition is only on H bus and E bus. The individual 4KV breakers for the RCP motors will trip open on undervoltage conditions and not on underfrequency. So in this, the crew would receive OHA J-37, 4KV GRP BUS UNDRFREQ, and all RCPs will remain running with the reactor trip breakers closed.

B. Incorrect. First part is correct. Second part is incorrect. Plausible if the operator makes an incorrect assumption that the 4KV Group Bus breakers open on underfrequency conditions. This is not true, the RCP breakers open on an undervoltage condition.

C. Incorrect. First part is incorrect. Plausible because the operator may incorrectly believe that the underfrequency on H and E Bus will trip open 21 and 22 RCP breakers, thus satisfying the 2/4 RC loop low flow reactor trip logic with the plant below P-8 (< 36%) but above P-7 (>10%). Second part is incorrect. Plausible because if the operator believes that the UF condition on H and E Buses satisfies the 1 out of two taken twice, resulting in all RCPs tripping.

D. Incorrect. First part is incorrect. Plausible because the operator may incorrectly believe that the underfrequency on H and E Bus will trip open 21 and 22 RCP breakers, thus satisfying the 2/4 RC loop low flow reactor trip logic with the plant below P-8 (< 36%) but above P-7 (>10%). Second part is incorrect. Plausible if the operator believes that only the RCP breakers on H and E Bus will trip open on a UF condition, which is not true.

Technical References:	Fluency List
Proposed References to be provided:	None
Learning Objective:	NOS05RXPROT-15, Objective 12.a. State the setpoints, coincidence, blocks and permissives for all Reactor Trips and Safety Injections actuations
Question Source:	Bank – Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.2 to 41.9 / 45.7 to 45.8
Level:	RO
Category/System:	07-Reactor Protection System
K/A #:	012 K1.09 Knowledge of the physical connections and/or cause-effect relationships between the RPS and the following systems: RCPS
Importance:	3.8
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked how the RCP system relates to the RPS system. In this case, the stem provides conditions (underfrequency and undervoltage) to determine if the applicant knows the trip logic associated with RCP underfrequency.

Additional Reference:

VI. REACTOR TRIP SETPOINTS

SLIDE-17; ELO-12 (Operator Fundamental – Closely Monitor Plant Indications)

Name	Setpoint	Coincidence	Permissive
Manual	Manual	1/2 Switches	None
SI	Any SI	1 key switch ("S" signal) on either train	None
SR High Flux	1.0×10^5 cps	1/2 SR Detectors	P-6 allows block
IR High Flux	25 %	1/2 IR Detectors	P-10 allows block
PR High Flux LO SP	25%	2/4 PR Detectors	P-10 allows block
PR High Flux Hi SP	109%	2/4 PR Detectors	None
PR High Positive Rate	+5 % of RTP w/2 sec TC	2/4 PR Detectors	None
OTΔT	Variable[1]	2/4 ΔT's	None
OPΔT	Variable[2]	2/4 ΔT's	None
SG LO-LO Level	14 %	2/3 levels on 1/4 SG's	None
RCP Breaker Open	2 Breakers Open	2 RCP breakers	> P-7
Loss of Flow	<90%	2/3 on 1/4 RC Loops	> P-8
Loss of Flow	<90%	2/3 on 2/4 RC Loops	> P-7
RCP Undervoltage	<70% for >0.1 sec	1/2 on Group Buses H or E (AND) 1/2 on Group Buses F or G	> P-7
RCP Underfrequency	<56.5Hz, >0.1 sec	Same as above	> P-7
PZR Pressure High	>2385 psig	2/4 PZR Press.Channels	None
PZR Pressure Low	<1865 psig	2/4 PZR Press.Channels (rate compensated)	> P-7
PZR Level High	>92%	2/3 PZR Level Channels	> P-7
Turbine Trip	<45 psig ASOP[3] or 4/4 stop valves closed	2/3 channels for <45 psig or Stop valves 15% closed from open limit	> P-9

[1] $OT\Delta T_{setpoint} = \Delta T_0 \{ K1 - K_2 [(1 + \tau_1 s) \div (1 + \tau_2 s)] (T - T') + K_3 (P - P') - f_1(\Delta\phi)\}$

[2] $OP\ \Delta T_{setpoint} = \Delta T_0 \{K_4 - K_5 [(\tau_3 s \div (1 + \tau_3 s)] T - K_6 (T - T'') - f_2(\Delta\phi)\}$

37 **02-033** **Points: 1.00**

Given:

- Salem Unit 1 experienced a reactor trip and an SI actuation due to a LOCA
- The crew is performing 1-EOP-LOCA-1, Loss of Reactor or Secondary Coolant
- SI has been RESET
- SECs have been RESET
- ALL EDGs have been stopped from the control room

Then:

- Loss of Off-site Power occurs

Which of the following describes the expected response of the EDGs and **safeguard loads**?

A. EDGs require manual start and some safeguard loads need to be manually started.

B. EDGs require manual start and SECs automatically sequence ALL safeguard loads.

C. EDGs automatically start and SECs automatically sequence ALL safeguard loads.

D. EDGs automatically start and some safeguard loads need to be manually started.

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that since the EDGs were stopped and the SECs reset, that the EDGs will not auto start.

B. Incorrect. Plausible because the SEC will automatically sequence loads. Incorrect in that the SEC in Mode 2 will only start the charging pumps, the other safeguard loads need to be started manually.

C. Incorrect. Plausible because the SEC will automatically sequence loads. Incorrect in that the SEC in Mode 2 will only start the charging pumps, the other safeguard loads need to be started manually.

D. Correct. After the SI is reset and SECs are reset, the SI signal to the SEC logic are cleared and the SECs are ready for automatic actuation. The EDGs will automatically start on a Blackout condition (Mode 2) and the SECs will automatically sequence loads for Blackout only. The only safeguards load started during Mode 2 are the charging pumps. The crew will be required to start the other safeguard loads (e.g, SI pumps, RHR Pumps, and CFCUs) since the SI and SECs were reset prior to the LOOP.

Technical References:	NOS05SEC000-09
Proposed References to be provided:	None
Learning Objective:	NOS05SEC000-09, Objective 4. Describe in detail each Mode of Safeguards Equipment Control System operation, including setpoints for automatic actuation
Question Source:	Bank – Commanche Peak 2020 RO20
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.5
Level:	RO
Category/System:	02-Engineered Safety Features Actuation System
K/A #:	013 A1.14 Ability to predict and/or monitor changes in parameters associated with operating the ESFAS system including: EDG
Importance:	3.8
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked to predict how the EDGs will respond to a LOOP after the SI signal and SECs have been reset.

38 **05-022** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- 21, 22, 23 and 24 CFCUs are running in HIGH speed with 25 CFCU in standby
- An inadvertent SI actuation occurs

Which of the following completes both statements?

1. Following SI actuation, all CFCUs will start in LOW speed __(1)__.
2. To restore CFCUs to HIGH speed operation, __(2)__ must be RESET.

A. (1) after 20 seconds
(2) Only SI

B. (1) after 20 seconds
(2) SI and SECs

C. (1) immediately
(2) SI and SECs

D. (1) immediately
(2) Only SI

Answer: B

Explanation / Justification:

A. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because the operator may believe that with an SI only, that only resetting is required. Incorrect in that the SECs are required to be reset as well.

B. Correct. Inadvertent SI actuation will actuate the SECs in MODE 1 operation. The CFCUs running in high speed will be tripped and started 20 seconds later in low speed. The CFCU in standby will start 20 seconds later following the SI actuation. In order to regain manual control of the CFCUs to return them to high speed operation, the operator will need to reset both trains of SI and reset all SECs. Step 1 of 2-EOP-TRIP-3, SI Termination, will accomplish this.

C. Incorrect. Part 1 is incorrect. Plausible because the operator may believe that with no loss of off-site power present, then the CFCUs can immediately start since no diesel overloading issue. Incorrect in that a time delay is still required to allow the CFCUs to coast down. Part 2 is correct.

D. Incorrect. Incorrect. Part 1 is incorrect. Plausible because the operator may believe that with no loss of off-site power present, then the CFCUs can immediately start since no diesel overloading issue. Incorrect in that a time delay is still required to allow the CFCUs to coast down. Part 2 is incorrect. Plausible because the operator may believe that with an SI only, that only resetting is required. Incorrect in that the SECs are required to be reset as well.

Technical References:	NOS05SEC000-09, 2-EOP-TRIP-3 Sht. 1(R41)
Proposed References to be provided:	None
Learning Objective:	NOS05SEC000-09, Objective 9. Describe how to reset the Safeguards Equipment Control System after a Mode Operation, and what happens when the Safeguards Equipment Control System is reset
Question Source:	Modified – Braidwood 2016 NRC RO20
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.6
Level:	RO
Category/System:	05-Containment Cooling System
K/A #:	022 A2.07 Ability to (a) predict the impacts of the following on the Containment Cooling System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: ESFAS actuation
Importance:	4.0
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked to how the ESFAS actuation affects the status of the CFCUs and what action(s) is necessary to regain control of the CFCU to restore to normal operation.

Parent Question: Braidwood 2016 NRC RO20

Modified question based on adding Part 2 to ask what actions are required to restore CFCUs to high speed operation.

20 **ID: 1267989** **Points: 1.00**

Given:

- Unit 1 is at full power.
- 1A, 1B and 1C RCFCs are running in high speed with the 1D RCFC in standby.
- A reactor trip and Safety Injection occur due to high containment pressure.

After the SI occurs, what is the response of the RCFCs?

A. ALL RCFCs start immediately in low speed.

B. 1A, 1B and 1C RCFCs start immediately in low speed, and the 1D RCFC starts 20 seconds later in low speed.

C. 1A, 1B and 1C RCFCs start 20 seconds later in low speed, and the 1D RCFC starts immediately in low speed.

D. ALL RCFCs start 20 seconds later in low speed.

Answer: D

39 **05-023** **Points: 1.00**

Given:

- Salem Unit 2 experienced a reactor trip and safety injection actuation due to a LOCA
- 23 CFCU is C/T for maintenance
- 2B 4KV Vital Bus de-energized on Bus Differential protection

Which of the following identifies the status of the CFCUs?

A. 22 and 25 are running Only.

B. 22 and 24 are running Only.

C. 21 and 24 are running Only.

D. 21 and 25 are running Only.

Answer: D

Explanation / Justification:

A. Incorrect. Plausible if the operator incorrectly recalls the power supplies to the CFCUs and believes that 23 and 25 CFCUs are supplied from A and C bus. Incorrect in that only 22 and 24 CFCUs will not be running due to B bus de-energized.

B. Incorrect. Plausible if the operator incorrectly recalls the power supplies to the CFCUs and believes that 22 and 24 CFCUs are supplied from A and C bus. Incorrect in that 22 and 24 CFCUs will not be running due to B bus de-energized.

C. Incorrect. Plausible if the operator incorrectly recalls the power supplies to the CFCUs and believes that 21 is supplied from A bus and 22 and 24 CFCUs are supplied from C bus. Incorrect in that 22 and 24 CFCUs will not be running due to B bus de-energized

D. Correct. The CFCUs are supplied from the following 4KV Vital Busses; 21 – A bus, 22 – B bus, 23 – C bus, 24 – B bus, and 25 – C bus. With 2B 4KV bus de-energized and 23 CFCU C/T, then only 21 and 25 CFCUs are running.

Technical References:	NOS05CONTMT-17
Proposed References to be provided:	None
Learning Objective:	NOS05CONTMT-17, Objective 4. State the power supply to the following Containment and Containment Support Systems components, including voltage level and 1E/Non 1E
Question Source:	Bank – Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.7
Level:	RO
Category/System:	05-Containment Cooling System
K/A #:	022 K2.01 Knowledge of the electrical power supplies to the following: Containment Cooling System CCS fans
Importance:	3.6
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked which CFCUs will be running following a loss of 2B 4KV bus.

40 **05-024** **Points: 1.00**

Given:

- Salem Unit 2 experienced a LOCA
- Containment pressure is 10 psig and slowly rising
- ALL available safeguards equipment have started with the exception that 2B SEC failed to actuate
- SECs have NOT been RESET

Based on the conditions above, which of the following completes the statement?

__(1)__ containment spray pump(s) is/are still available from a Class 1E vital power source and __(2)__ automatically start when containment pressure exceeds the containment pressure HI-HI setpoint.

A. (1) ONLY one (2) will

B. (1) ONLY one (2) will NOT

C. (1) BOTH (2) will

D. (1) BOTH (2) will NOT

Answer: C

Explanation / Justification:

A. Incorrect. Part 1 is incorrect. Plausible if the operator incorrectly recalls the power supplies to the CS pumps and believes that when the SEC is at the end of the sequence the CS pumps will not start when a containment pressure HI-HI signal is present. Incorrect in that at the end of sequence and with the SEC not reset, the SEC will auto start a CS pump when a containment pressure HI-HI signal is present. Part 2 is correct.

B. Incorrect. Part 1 and Part 2 are incorrect. Plausible if the operator incorrectly recalls the power supplies to the CS pumps and believes that when the SEC is at the end of the sequence the CS pumps will not start when a containment pressure HI-HI signal is present. Incorrect in that at the end of sequence and with the SEC not reset, the SEC will auto start a CS pump when a containment pressure HI-HI signal is present.

C. Correct. Both containment spray pumps will still be powered from the 2A and 2C 4KV Vital Busses with the 2B SEC failing to actuate. The 2A and 2C SECs have actuated in MODE 1 due to a LOCA only and with the sequence complete and the SECs not reset, a containment pressure HI-HI signal will automatically start the 21 and 22 CS pumps since the SECs are still waiting for a CS HI-HI signal when sequence is complete.

D. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because the SECs sequencer allows for two opportunities to start a CS pump. The operator may incorrectly recall this and believe that when the SEC sequence is complete, the operator will have to manually start a CS pump. Incorrect in that at the end of sequence and with the SEC not reset, the SEC will auto start a CS pump when a containment pressure HI-HI signal is present.

Technical References:	NOS05SEC000-09
Proposed References to be provided:	None
Learning Objective:	NOS05SEC000-09, Objective 4. Describe in detail each Mode of Safeguards Equipment Control System operation, including setpoints for automatic actuation
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7
Level:	RO
Category/System:	05-Containment Spray System
K/A #:	026 K2.01 Knowledge of electrical power supplies to the following: Containment Spray System containment spray pumps
Importance:	3.9
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked which containment spray pumps are still available to start given a loss of the 2B SEC and will the containment spray pumps auto start when the containment spray HI-HI signal is exceeded. In this situation, the candidate's knowledge of electrical power supplies for the Containment Spray pump is tested by the knowledge and impact to the CS pumps based on the available SECs. The A and C SECs, supplied by A and C 115 Vital Instrument Power buses, starts the CS pump on the respective 4KV buses (i.e., 'A' SEC starts 21 CS pump supplied by 'A' 4KV bus and 'C' SEC starts 22 CS Pump supplied by 'C' 4KV bus)

Additional References: NOS05SEC000

LESSON NAME:	SAFEGUARDS EQUIPMENT CONTROL SYSTEM	
	NOS05SEC000-09	02/18/21

g. Containment Spray Pump Sequencing

1) If the SSPS Containment HI-HI Pressure signal is not present when the SEC initially tries to start the Spray Pumps, the SEC contact will re-open

2) Once the SEC has completed the last step of its loading sequence, the CS Pump start contact is re-closed

a) If Hi-Hi Containment Pressure occurs the CS Pumps will auto start

b) If the SEC has been reset, the CS Pumps will NOT respond to a HI-HI Containment Pressure until the SEC is again actuated

c) CS Pumps do not have circuit breaker "a" contacts in the "Sequence Failure" or "Loading Complete" circuits. Sequence Failure will NOT occur as a result of a CS Pump not starting until the SSPS HI-HI Pressure signal is received. CS pumps also do not have individual component status lights on RP-4 Status Panel

41 **04-073** **Points: 1.00**

Given:

- Salem Unit 2 reactor tripped and actuation of Main Steam Line Isolation occurred due to a steamline break
- ONLY the 23 AFW Turbine-Driven feedpump is running supplying all SGs
- ALL SG pressures are 900 psig and lowering rapidly

Which of the following completes the statement concerning the effect on the AFW system (assuming no operator action)?

When SG pressures lowers to 700 psig, 23 AFW turbine speed will __(1)__ and feedflow will ___(2)___.

A. (1) lower (2) lower

B. (1) remain constant (2) rise

C. (1) remain constant (2) remain constant

D. (1) lower (2) rise

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that as SG pressure lowers so will the turbine speed and feedflow.

B. Correct. 23 AFW Turbine Driven Pump is designed to operate with SG pressures at 100 psig. As SG pressures lower, the 23 AFW governor will throttle to maintain relative speed of the turbine constant. However, as SG pressure lowers, the D/P across the feedpump discharge and the SG will increase resulting in more AFW flow to the SGs.

C. Incorrect. Plausible because turbine speed will be maintain relatively constant and the operator may believe that feedflow will also remain constant. Incorrect in that feedflow will rise due to the lowering SG pressure.

D. Incorrect. Plausible because feedflow will rise as SG pressure lowers.

Technical References:	NOS05AFW000-18
Proposed References to be provided:	None
Learning Objective:	NOS05AFW000-18, Objective 15.c. Given plant conditions, relate the Auxiliary Feedwater System with the following, Main Steam
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 / 45.4
Level:	RO
Category/System:	04-Main and Reheat Steam System
K/A #:	039 K3.03 Knowledge of the effect that a loss or malfunction of the Main and Reheat Steam System will have on the following systems or system parameters: AFW system
Importance:	3.3
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked knowledge of the effects of the turbine driven AFW pump when SG pressure lowers due to a faulted SGs and how feedflow will also behave with lowering SG pressure.

Additional Reference:

c. Turbine

1) Single stage, atmospheric condensing, Terry Turbine

a) Inlet steam conditions

(1) Normal - 1045 psig, 552°F

(2) Maximum - 1045 psig, 552°F

(3) Minimum - 85 psig, 327°F

42 **04-074** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- 21BF19, MFW Regulating Valve, fails fully OPEN

Which of the following completes the statement regarding how the plant will respond?

The reactor will trip and the crew will enter 2-EOP-TRIP-1, Reactor Trip or Safety Injection, when high level on SG NR levels reach __(1)__ followed by a __(2)__.

	(1)	(2)
A.	67%	Feedwater Isolation
B.	67%	Feedwater Interlock ONLY
C.	92%	Feedwater Isolation
D.	92%	Feedwater Interlock ONLY

Answer: A

Explanation / Justification:

A. Correct. Failed open BF19 will result in a Main Turbine Trip on high SG level followed by a reactor trip. The SG High Level setpoint is 2 out of 3 SG NR levels ≥ 67% on any one SG (P-14). P-14 will trip the main turbine that will cause an auto reactor trip and actuate a FW Isolation to close all BF13s, BF19s, BF40s, trip both SGFPs, and trips Main Turbine. FW interlock only closes the BF19s and BF40s.

B. Incorrect. Plausible because a reactor trip will actuate a FW Interlock. Incorrect in that the SG high level (P-14) signal actuate a FW Isolation.

C. Incorrect. Plausible because this is the reactor trip setpoint for PZR high level.

D. Incorrect. Plausible because this is the reactor trip setpoint for PZR high level. Incorrect in that the P-14 setpoint is 67% and a FW Isolation occurs.

Technical References:	Drawing ESF-1
Proposed References to be provided:	None
Learning Objective:	NOS05RXPROT-15, Objective 12.a. State the setpoints, coincidence, blocks and permissives for all Reactor Trips and Safety Injections actuations
Question Source:	Bank - Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.7 / 45.8
Level:	RO
Category/System:	04-Mai Feedwater System / Emergency Procedures/Plan
K/A #:	059 G2.4.2 Knowledge of system setpoints, interlocks, and automatic actions associated with emergency and abnormal operating procedure entry conditions
Importance:	4.5
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked the setpoint for the SG High level Main Turbine/Reactor Trip and the auto action for FW Isolation.

43 **04-075** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- Instrument air supply line to 2MS132, 23 AFW Turbine Steam Inlet Valve, ruptures

Which of the following completes both statements?

1. What is the status of 23 AFW Pump?
2. What action should the operator take FIRST?

A. (1) Running
(2) Reduce reactor power to prevent overpower condition per S2.OP-IO.ZZ-0004, Power Operation

B. (1) Running
(2) Take manual control of the Main Feedwater Regulating valves to prevent SG overfill per S2.OP-AB.CA-0001, Loss of Control Air.

C. (1) NOT Running
(2) Enter Tech Spec LCO 3.7.1.2, Auxiliary Feedwater System for INOPERABLE AFW Pump.

D. (1) NOT Running
(2) Fail open the 2MS132 by using the handwheel on the valve actuator per S2.OP-AB.CA-0001, Loss of Control Air.

Answer: A

Explanation / Justification:

A. Correct. Air supply line rupturing on the 2MS132 will cause the AFW steam inlet valve to fail open. This will result in 23 AFW running and injecting cold feedwater into the SGs. As a result, reactor power will increase above the 100% RTP and the operator should reduce reactor power per S2.OP-IO.ZZ-0004, Power Operation. Per S2.OP-IO.ZZ-0004, the maximum power level is 3459 MWt (100% RTP).

B. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because with 23 AFW running additional feedwater is being supplied to all the SGs. Incorrect in that with digital feedwater system in Automatic, the MFW Regulating Valves will automatically adjust demand to maintain SG NR levels within program band.

C. Incorrect. Plausible because the operator may incorrectly recall how the valve fails on loss of air and believe the 2MS132 fails closed thereby preventing the start of 23 AFW pump. Entering TS LCO 3.7.1.2 is plausible because this LCO is applicable for the 23 AFW pump condition.

D. Incorrect. Plausible because the operator may incorrectly recall how the valve fails on loss of air and believe the 2MS132 fails closed thereby preventing the start of 23 AFW pump. Failing open the 2MS132 is plausible because many AOVs can be locally operated.

Technical References:	NOS05AFW000-18, S2.OP-IO.ZZ-0004 (R88)
Proposed References to be provided:	None
Learning Objective:	NOS05AFW000-18, Objective 16. Given an Auxiliary Feedwater System failure, predict the effect of the Auxiliary Feedwater System failure on the following: Steam Generators
Question Source:	Bank – Salem
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.6
Level:	RO
Category/System:	04-Auxiliary Emergency Feedwater System
K/A #:	061 A2.07 Ability to (a) predict the impacts of the following on the Auxiliary Emergency Feedwater System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Air operated valve, solenoid-operated valve, or motor-operated valve
Importance:	4.0
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the applicant is being asked how the AFW AOV will fail on loss of air and what action should the operator take based on the failure of the AOV.

44 **04-076** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

Then:

- The Primary Equipment Operator reports a significant air leak on the air supply line at the connection to the 2DR6, AFW Tank Makeup, valve actuator

Which of the following describes the expected impact to the Auxiliary Feedwater System (assuming the leak is causing a reduction in air pressure to the actuator) and operator response, if any?

AFW storage tank level will…

A. rise and overflow; local/manual control is NOT available.

B. rise and overflow; local/manual control is available.

C. remain the same; local/manual control is available.

D. remain the same; local/manual control is NOT available.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because not all air operated valves can be manually operated in the field.

B. Correct. The 2DR6 will fail to the open position on loss of air. This will result in the AFWST level rising and potentially overflowing the tank. Per S2.OP-AB.CA-0001, Attachment 2, the 2DR6 can be manually operated from the field to control tank level.

C. Incorrect. Plausible because the operator may believe that the 2DR6 failing open would result in tank overflow, thus its failure position would be closed to prevent this (air to open actuator).

D. Incorrect. Plausible because the operator may believe that the 2DR6 failing open would result in tank overflow, thus its failure position would be closed to prevent this (air to open actuator).

Technical References:	NOS05AFW000-18, S2.OP-AB.CA-0001 (R22)
Proposed References to be provided:	None
Learning Objective:	NOS05AFW000-18, Objective 15.f. Given plant conditions, relate the Auxiliary Feedwater System with the following: Demineralized Water System
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.3
Level:	RO
Category/System:	04-Auxiliary Feedwater System / Components
K/A #:	191001 K1.04 The failed-valve positions for different operators (open, closed, and as-is positions spring loaded valves; hydraulic, pneumatically controlled valves; electric motor-driven valves)
Importance:	3.2
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the question is asking how the loss of air to the 2DR6 AOV will be impacted and the effect on the AFWST level.

Additional Reference: S2.OP-AB.CA-0001, Attachment 2

AUXILIARY FEEDWATER SYSTEM

Valve	Fails	Description	Hdr/Bkup
21-24AF11	FO ♦	23AFP Disch to 21-24SG	2B/2A
21-24AF21	FO ♦	AFP to 21-24SG	2B/2A
21-23AF52	FC ♦	AFP Suct XConn	2B/2A
21-22AF40	FO ♦	21-22AF Pump Recirc	2B/2A
2MS53	FHSS ▽	23AFP Turbine Governor	2B/2A
2MS132	FO	23AFP Turbine Steam Inlet	2B/2A
2DR6	FO ♦	Supply to AFW	2B

♦ Can be manually operated

▽ Fails at High Speed Stop / Can be manually operated

Salem 2 Page 38 of 88 Rev. 22

45 **06-029** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- 23 Charging Pump is in service
- 22 PZR Backup Heaters are ON
- A Overcurrent fault on the 2A 460V Motor Control Center (MCC) causes the 4160V feeder breaker 2A4D to trip open, de-energizing the 2A 460V and 230V MCCs

Per S2.OP-AB.460-0001, Loss of 2A 460 V Bus, which of the following describes the effect on plant operations and the required action(s)?

A. Loss of 2B1 125 VDC Battery Charger; place 2B2 125 VDC Battery Charger in service.

B. Loss of charging flow; start a centrifugal charging pump and restore PZR level.

C. Loss of all IRPI indications; transfer to emergency power source.

D. Loss of 22 PZR Backup Heaters; re-energize from emergency power bus.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the operator may incorrectly recall which 230 V bus supplies the normal and backup battery chargers and believe that the 2B1 125VDC is supplied from 2A 230 V bus. Incorrect in the2B1 125VDC Battery Charger is supplied from the 2B 230 V bus and the 2B2 125VDC battery charger is supplied from the 2A 230 V bus.

B. Correct. 23 charging pump is supplied from the 2A 460 V bus. An overcurrent condition on the 2A 460 V MCC/bus will cause the breakers associated on the 2A 460 and 230 V MCCs to trip open on overcurrent and de-energize the 2A 460V bus. This will result in loss of 23 charging pump and letdown isolation when the 23 charging pump breaker opens. Per S2.OP-AB.460-0001, Loss of 2A 460/230V Vital Bus, the operator will start a centrifugal charging pump and then re-establish letdown.

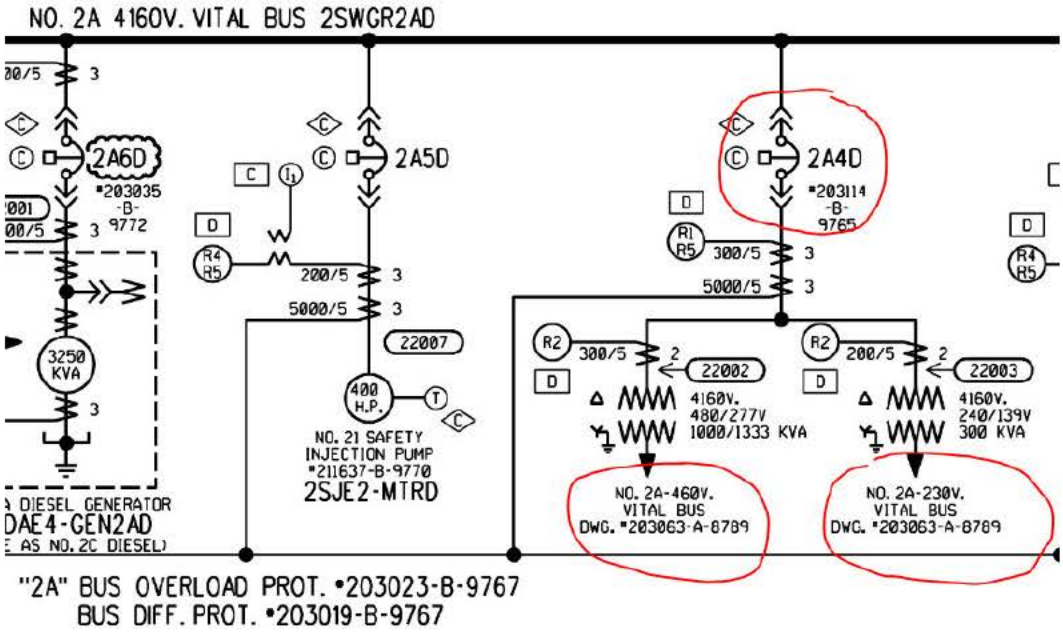
C. Incorrect. Plausible because the operator may incorrectly recall which 230 V bus supplies the normal and emergency power to the IRPI indications and believe that the 2A 230V bus is the normal supply resulting in a loss of IRPI indications. Incorrect in that the normal power supply is from 2B 460V and the 2A 230V bus supplies the emergency power.

D. Incorrect. Plausible because the operator may incorrectly recall which 460 V bus supplies the normal and emergency power to the Pressurizer backup heaters and believe that the 2A 460V bus is the normal supply resulting in a loss of PZR backup heaters Group 22. Incorrect in that the normal power supply is from the EP 460V bus and the 2A 460V bus supplies the emergency backup.

Technical References:	203061 (R39), 601390 (R25), S2.OP-AB.460-0001 (R9)
Proposed References to be provided:	None
Learning Objective:	NOS05460VAC-09, Objective 4.c. Describe the function of the following components and how their normal and abnormal operation affects the 460/230VAC Electrical System: 4KV/460V Group Transformers
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.6
Level:	RO
Category/System:	06-AC Electrical Distribution System
K/A #:	062 A2.25 Ability to (a) predict the impacts of the following on the AC Electrical Distribution System and (b) based on those predications, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Fault on a motor control center.
Importance:	3.3
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked to predict the impacts on plant operations following an overcurrent condition on the 2A 460V bus that cause the upstream 4160V feeder breaker (2A4D) to trip open resulting in a loss of 2A 460 and 230V busses. The operator will recall knowledge of the major impacts from this loss on the major action from the abnormal procedure.

Additional References: 203061



46 **06-030** **Points: 1.00**

Given:

- Salem Unit 2 was at 100% power
- During control room log taking, the operator observes the 2A 125 VDC Bus ground ohmmeter reading 40K Ω (ohms)

Based on the above condition, which of the following completes both statements?

1. The reading indicates that a ground on the bus __(1)__ detected.
2. If a ground is detected, S2.OP-SO.125-0004, 125VDC Ground Detection, will __(2)__.

A. (1) is NOT
(2) open all breakers in the panel, then reclose them one at a time, if no change to ground readings then continue

B. (1) is NOT
(2) open one breaker at a time, evaluate ground readings, leave open if ground clears

C. (1) is
(2) open one breaker at a time, evaluate ground readings, leave open if ground clears

D. (1) is
(2) open all breakers in the panel, then reclose them one at a time, if no change to ground readings then continue

Answer: C

Explanation / Justification:

A. Incorrect. First part is incorrect. Plausible because the operator can incorrectly recall and confuse which resistance reading indicates a ground present and believe that the 0 ohms indicates no grounds. Incorrect in that resistance reading less than 50K ohms is a ground. Second part is incorrect. Plausible again if the operator confuses how to interpret resistance readings and believes a lower resistance is good.

B. Incorrect. First part is incorrect. Plausible because the operator can incorrectly recall and confuse which resistance reading indicates a ground present and believe that the 0 ohms indicates no grounds. Incorrect in that resistance reading less than 50K ohms is a ground. Second part is correct.

C. Correct. Per S2.OP-SO.125-0004, a resistance reading of 50K ohms or less indicates a ground is detected. A resistance of greater than 50K ohms is normal indicating no grounds are present. Therefore, a ground reading of 0 ohms indicates a hard ground is present. Per S2.OP-SO.125.0004, breakers on the bus will be opened one at a time until the resistance indicates greater than 50K ohms.

D. Incorrect. First part is correct. Second part is incorrect. Plausible because the operator can incorrectly recall and confuse which resistance reading indicates a ground present and believe that the 0 ohms indicates no grounds. Incorrect in that resistance reading less than 50K ohms is a ground.

Technical References:	NOS05DCELECT-11
Proposed References to be provided:	None
Learning Objective:	NOS05DCELEC-11, Objective 8.b. Identify and describe the Control Room controls, indications, and alarms associated with the DC Electrical System, including: The function of each DC Electrical System Control Room control and indication
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.2 to 41.9 / 45.7 to 45.8
Level:	RO
Category/System:	06-DC Electrical Distribution System
K/A #:	063 K1.01 Knowledge of the physical connections and/or cause and effect relationships between the DC Electrical Distribution System and the following systems: Ground detection system.
Importance:	2.6
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked how the ground detection system relates to the DC bus as far as the expected indications for a ground on the bus. Also, the candidate is asked how the actions in the procedure will determine the isolation of a ground.

47 **06-031** **Points: 1.00**

Which of the following completes both statements concerning the 125 VDC Electrical Distribution System?

1. The 125 VDC Distribution System design __(1)__ include a mechanical interlock when transferring from normal to the backup (emergency) bus.

2. Transferring 125 VDC distribution loads to the backup bus is permitted during __(2)__.

A. (1) does NOT
(2) ALL Modes

B. (1) does NOT
(2) Modes 5,6, or defueled Only

C. (1) does
(2) ALL Modes

D. (1) does
(2) Modes 5,6, or defueled Only

Answer: D

Explanation / Justification:

A. Incorrect. Part 1 is incorrect. Plausible because the operator may incorrectly recall that the DC busses do not have a mechanical interlock. Incorrect in that a mechanical interlock is part of the design. Part 2 is correct. Part 2 is incorrect. Plausible because the operator may misinterpret the questions and believe that the DC bus can be transferred during any modes. Incorrect in that paralleling DC bus will make the DC busses Inoperable during Modes 1-4.

B. Incorrect. Part 1 is incorrect. Plausible because the operator may incorrectly recall that the DC busses do not have a mechanical interlock. Incorrect in that a mechanical interlock is part of the design. Part 2 is correct.

C. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because the operator may misinterpret the questions and believe that the DC bus can be transferred during any modes. Incorrect in that paralleling DC bus will make the DC busses Inoperable during Modes 1-4.

D. Correct. The 125 VDC system does include a design feature that prevents paralleling the normal and backup busses. A mechanical interlock located in the distribution panel requires the normal supply breaker to be open and then the mechanical device is allowed to move allowing the closing of the backup or emergency power. Per procedure, transferring to the backup DC bus is only allowed during Modes 5, 6, or defueled.

Technical References:	NOS05DCELECT-11
Proposed References to be provided:	None
Learning Objective:	NOS05DCELECT-11, Objective 6.a. Outline the interlocks associated with the following DC Electrical System components: Normal and Emergency feeds to loads on125VDC buses
Question Source:	Bank – Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.7
Level:	RO
Category/System:	06-DC Electrical Distribution System
K/A #:	063 K4.01 Knowledge of DC Electrical Distribution System design features and/or interlocks that provide for the following: Manual/automatic transfers of control.
Importance:	3.4
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked knowledge of 125VDC design feature with regards to a mechanical interlock used for transferring from normal to emergency buses.

48 **06-032** **Points: 1.00**

What are the power supplies to the 2A EDG starting air compressors 21A and 21B?

	21A	**21B**
A.	2A 230V Vital Bus	2A 230V Vital Bus
B.	2A 230V Vital Bus	2B 230V Vital Bus
C.	2A 460V Vital Bus	2A 460V Vital Bus
D.	2A 460V Vital Bus	2B 460V Vital Bus

Answer: A

Explanation / Justification:

A. Correct. Each EDG two starting air compressors are supplied from its associated bus 230 V vital bus. The 21A and 21B starting air compressors are both powered from 2A 230V Vital Bus.

B. Incorrect. Plausible because the operator may believe that the 21B starting air compressor is powered from the other EDG 230V vital bus. Incorrect in that both starting air compressors are powered from the same 2A 230V vital power.

C. Incorrect. Plausible because the operator may incorrectly recall and believe that the starting air compressors are powered from 2A 460V vital bus. Incorrect in that both starting air compressors are powered from the same 2A 230V vital power.

D. Incorrect. Plausible because the operator may believe that the 21B starting air compressor is powered from the other EDG 460V vital bus. Incorrect in that both starting air compressors are powered from the same 2A 230V vital power.

Technical References:	NOS05EDG000-15
Proposed References to be provided:	None
Learning Objective:	NOS05EDG000-15, Objective 5.e. State the power supply to the following Emergency Diesel Generator components: Starting Air Compressors
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.7
Level:	RO
Category/System:	06-Emergency Diesel Generator System
K/A #:	064 K2.01 Knowledge of electrical power supplies to the following Emergency Diesel Generator System: Starting air compressor
Importance:	3.0
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked the power supplies to the EDG air compressors.

49 **07-026** **Points: 1.00**

Given:

- Salem Unit 1 is at 100% power
- The PO needs to perform a check source on the 1R19A, SGBD Radiation Monitor

Per S1.OP-ST.RM-0001, Radiation Monitors – Check Source, which of the following completes both statements?

1. When the operator pushes the check source pushbutton, the radiation monitor will respond by ___(1)__.
2. During the check source, actuation of interlocks __(2)__ expected.

A. (1) a rise in counts until the check source pushbutton is released
(2) are

B. (1) a rise in counts until the check source pushbutton is released
(2) are NOT

C. (1) a rise in counts then a return to original reading
(2) are NOT

D. (1) a rise in counts then a return to original reading
(2) are

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because other radiation monitors do perform like this. Incorrect in that this radiation monitor check source the counts rise then fall and interlocks are not expected during check source.

B. Incorrect. Plausible because other radiation monitors do perform like this. Incorrect in that this radiation monitor check source the counts rise then fall and interlocks are not expected during check source.

C. Correct. When a check source is performed on this radiation monitor, the counts will rise then return to its previous reading. During check source it is not expected for interlocks to actuate.

D. Incorrect. Plausible because some other radiation monitors interlocks could occur during check source.

Technical References:	NOS05RMS000-20, S1.OP-ST.RM-0001(R10)
Proposed References to be provided:	None
Learning Objective:	NOS05RMS000-20, Objective 3. Describe how the following components impact the Radiation Monitoring System during normal and abnormal conditions: R19A, B, C, & D, Steam Generator Blowdown Liquid Monitors
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.7 / 45.5 to 45.8
Level:	RO
Category/System:	07-Process Radiation Monitor
K/A #:	073 A4.02 Ability to manually operate and/or monitor the Process Radiation Monitor in the control room: RMS control panel.
Importance:	3.6
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the question is asking what the expected response at the radiation monitor panel for the 1R19 during check source operation.

50 **04-077** **Points: 1.00**

Given:

- Salem Unit 1 is at 100% power
- Three (3) Service Water (SW) Pumps are in service
- Four (4) CFCUs are running in HIGH speed
- SW header pressures are 108 psig

Then:

- The PO is directed to place one running CFCU in LOW speed for surveillance testing

Which of the following completes the statement concerning the response on the Service Water System (assume three SW pumps remain running at all times)?

As compared to the SW header pressure while operating in HIGH speed, the SW header pressure will be __(1)__ when operating in LOW speed and SW flow rate through the CFCU will be __(2)___.

A. (1) lower
(2) the same

B. (1) lower
(2) higher

C. (1) the same
(2) the same

D. (1) the same
(2) higher

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because the SW flow through the CFCUs is the same for high or low speed operation. Correct in that the SW header will be relatively the same.

B. Incorrect. Plausible because prior operation of the CFCUs in low speed use to have a higher flow rate. Incorrect in that orifices have been installed to maintain constant flow rates in either speed.

C. Correct. The SW flow rates through the CFCUs in High and Low speed are the same, therefore, the SW header pressure will also be the same when operating in High and Low speed.

D. Incorrect. Plausible because prior operation of the CFCUs in low speed use to have a higher flow rate. Incorrect in that orifices have been installed to maintain constant flow rates in either speed.

Technical References:	NOS05CONTMT-17
Proposed References to be provided:	None
Learning Objective:	NOS05CONTMT-17, Objective 14. Given plant conditions, relate the Containment and Containment Support Systems with the following: Service Water
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.5
Level:	RO
Category/System:	04-Service Water System
K/A #:	076 A1.03 Ability to predict and/or monitor changes in parameters associated with operation of the Service Water System: SWS header pressure
Importance:	3.3
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the question is asking the applicant how the SW header pressure changes when the CFCU is swapped from High to Low speed.

51 **04-078** **Points: 1.00**

Given:

- Salem Unit 2 is in MODE 4
- 21 RHR Pump is in service providing shutdown cooling through 21 RHR HX
- RCS pressure is 300 psig
- 21RH18, RHR HX Flow Cont. Valve, and 2RH20, RHR HX Bypass Valve, are being throttled to maintain a RCS cooldown rate of 5°F per hour

Then:

- Instrument air supply line to the actuator of 21SW122, SW Inlet Control Valve, to CCW HX, severs

Assuming NO operator action, which of the following completes both statements?

1. RCS cooldown rate will ___(1)__.
2. The 21SW122 __(2)__ be manually operated.

A. (1) rise
(2) can NOT

B. (1) rise
(2) can

C. (1) lower
(2) can

D. (1) lower
(2) can NOT

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the operator may incorrectly recall if this valve can be manually operated from the field since not all AOVs can be manually operated in the field.

B. Correct. 21SW122 fails open on loss of air (fails closed on loss of 125VDC power). When the 21SW122 opens, SW flow will rise through the CCW HX causing CCW temperature to lower and therefore the outlet temperature of 21RHR HX to lower raising the cooldown rate. Per S2.OP-AB.CA-0001, the 21/22SW122 can be manually operated.

C. Incorrect. Plausible because the question is asking about the cooldown rate and the operator may confuse the concept that a lower temperature means a higher cooldown rate.

D. Incorrect. Plausible because the question is asking about the cooldown rate and the operator may confuse the concept that a lower temperature means a higher cooldown rate.

Technical References:	NOS05SW0NUC-13, S2.OP-AB.CA-0001 (R22)
Proposed References to be provided:	None
Learning Objective:	NOS05SW0NUC-13, Objective 4.f. Describe the function of the following components and how their normal and abnormal operation affects the Service Water – Nuclear Header System: CC HX SW122
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 / 45.6
Level:	RO
Category/System:	04-Service Water System
K/A #:	076 K3.05 Knowledge of the effect that a loss or malfunction of the Service Water System will have on the following systems or system parameters: RHRS
Importance:	3.6
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the question is asking the applicant how the SW122 service water control valve for the CCW HXs will impact the RHR system and the RCS cooldown rate..

52 **08-022** **Points: 1.00**

Given:

- Salem Unit 2 experienced a Safety Injection
- The crew enters 2-EOP-TRIP-1, Reactor Trip or Safety Injection
- Phase A signal failed to actuate on BOTH trains

Which of the following completes the statement concerning instrument air to containment?

1. 21 and 22 CA330s, Instrument Air Containment Isolation, valves are currently in the __(1)__ position.
2. MINIMUM requirement per 2-EOP-TRIP-1 for isolating instrument air to containment requires __(2)__ CA330(s) to be closed.

A. (1) closed
(2) both

B. (1) closed
(2) ONLY one

C. (1) open
(2) ONLY one

D. (1) open
(2) both

Answer: D

Explanation / Justification:

A. Incorrect Part 1 is incorrect. Plausible because the operator may believe that the CA330s close on an SI signal. Incorrect in that each CA330s will close on a Phase A signal only. Part 2 is correct. Plausible because the operator may believe that the CA330s are in series and need both to close

B. Incorrect. Part 1 is incorrect. Plausible because the operator may believe that the CA330s close on an SI signal. Incorrect in that each CA330s will close on a Phase A signal only. Part 2 is incorrect. Plausible because the operator may believe that the CA330s are in series and only need one of the two to isolate containment as a minimum.

C. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because the operator may believe that the CA330s are in series and only need one of the two to isolate containment as a minimum.

D. Correct. CA330s are closed by a Phase A isolation signal. Although SI actuated, with both trains of Phase A not actuated, then the 21 and 22 CA330s will remain open. Both 21 and 22 CA330s are required to be closed to fully isolate air to containment. 21CA330s supplies 2A control air header and 22CA330s supplies 2B control air header to containment. Each instrument air supply line to containment is isolated by its respective CA330 and a check valve. It is for this reason that both CA330s need to be closed to ensure a failure of one of the two components will still isolate a pathway from containment. Backup air receivers inside containment ensure that the PZR PORVs have sufficient air to allow the PZR PORV to stroke during steps in the EOP Network (Inadvertent SI, SGTR, etc.)

Technical References:	NOS05CONAIR-13
Proposed References to be provided:	None
Learning Objective:	NOS05CONAIR-13, Objective 4. Describe how each impact the Control Air System during normal and abnormal conditions: CA Containment Isolation Valves
Question Source:	Modified Bank – Farley 2019 NRC RO14
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.7 / 45.7
Level:	RO
Category/System:	08-Instrument Air System
K/A #:	078 A3.05 Ability to monitor automatic features of the Instrument Air System, including: Isolation of instrument air to containment
Importance:	3.3
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the applicant is being asked how to verify the automatic feature (Phase A signal) closed the CA330s. In this case, with no Phase A signal present, then both CA330s will remain open.

Parent Question: Farley 2019 NRC RO14

Modified Part 2 of the question.

Question # 14

Given the following:

- Unit 3 experiences an SI from 100% power.
- 3-EOP-E-0, Reactor Trip or Safety Injection, is in progress.

Which one of the following completes the statements below?

CV-3-2803, Instrument Air Containment Isolation is expected to be in the __(1)__ position.

Instrument Air supply to containment is essential for operation of __(2)__ in subsequent steps in the EOP Network.

A. (1) OPEN
(2) Pressurizer Spray valves.

B. (1) OPEN
(2) HCV-3-121, Charging Flow To Regen Heat Exchanger.

C. (1) CLOSED
(2) Pressurizer Spray valves.

D. (1) CLOSED
(2) HCV-3-121, Charging Flow To Regen Heat Exchanger.

Answer Analysis

Containment Isolation Signal does not close the Instrument Air to Containment Isolation valve at PTN. As per the Design Basis. Also, the design basis states that the verification of IA to Containment is prioritized in the procedure to ensure equipment in subsequent steps can be operated. Pressurizer spray is needed to stabilize RCS pressure when SI Termination criteria is met.

A. (1) Correct
(2) Correct

B. (1) Correct
(2) Incorrect, plausible because the HCV-121 is required to be operated when restoring Charging flow per EOP-2 Attachment 2, however this valve gets Instrument Air from the Aux Building IA Header.

C. (1) Incorrect, Plausible since this is a containment isolation valve, however, it does not get a CIV signal and fails open.
(2) Correct

D. (1) Incorrect, see C(1)
(2) Incorrect. See (B(2)

53 **05-025** **Points: 1.00**

Which of the following automatically generated signals will isolate the Containment Vacuum Relief Unit from the containment atmosphere when in service?

1. Phase A
2. Safety Injection
3. Containment Vent Isolation
4. Containment Spray/Phase B

A. 1 and 2 Only

B. 2 and 3 Only

C. 2, 3, and 4 Only

D. 1, 3, and 4 Only

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because Phase A signals do close penetrations through containment. Incorrect in that the Phase A signal does not actuate a CVI signal.

B. Correct. The Containment Vacuum Relief System filter unit uses only roughing filters. Since air is being drawn into containment there is no need to filter the air for radioactive contaminants. The dampers (VC5 and VC6) will automatically close on an automatic Safety Injection or CVI signal. The CVI signal will be actuated from a Safety Injection. CVI signal can also be generated manually using either (1) CS/Phase B keyswitches, or (2) Containment Vent Operate pushbutton on the safeguards console. Containment Atmosphere Radiation Monitors 2R11A, 2R12A, and 2R12B either one in alarm will cause a CVI.

C. Incorrect. Plausible because manually actuating CS/Phase B will actuate a CVI signal. Incorrect in that the stem is asking about automatically generated signals only.

D. Incorrect. Plausible because Phase A and B do isolate containment penetrations. Incorrect in that these signals does not actuate a CVI.

Technical References:	NOS05CONTMT-17, 221057 (RPS logic)
Proposed References to be provided:	None
Learning Objective:	NOS05CONTMT-17, Objective 5. Outline the interlocks associated with the Containment Pressure/Vacuum Relief System containment isolation valves, VC-5 and VC-6
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.7
Level:	RO
Category/System:	05-Containment System
K/A #:	103 K4.01 Knowledge of Containment System design features and/or interlocks that provide for the following: Vacuum breaker protection
Importance:	3.2
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the question is asking the applicant knowledge of the design features of the containment vacuum relief system used to relief negative pressure in containment.

Additional Reference: NOS05CONTMT

c. Indications

1) The Pressure Relief Unit has local indication of:

a) Roughing Filter differential pressure (PL-7690)

b) HEPA Filter differential pressure (PL-7689)

c) Charcoal Filter differential pressure (PL-7688)

2) The Vacuum Relief Unit has local indication of Roughing Filter differential pressure (PL-7691)

3) Local position indication is provided for:

a) CP - VRS inside containment isolation valve VC-6

b) CP - VRS outside containment isolation valve VC-5

c) Pressure Relief Unit Outlet Damper CBV36

d) Vacuum Relief Unit Outlet Damper CBV35

Additional Reference:

ELO 8.b

3) Containment Ventilation Isolation (CVI)

a) Initiated by:

- Safety Injection actuation
- High Containment atmosphere particulate, Iodine, or gaseous activity or High Plant Vent Release Rate
  - R11A, R12A & B, and R41D High Radiation alarm will initiate a CVI
- Manual actuation from the Control Room of:
  - Containment Ventilation Actuation
  - Phase B Containment Isolation/Containment Spray Actuation

b) The CVI signal will close:

- VC1, VC4, VC5, & VC6
- Note: VC2 & VC3 have been replaced by blind flanges VCF2 & VCF3. These flanges are the containment boundary in modes 1 through 4. They are removed to support containment purge in modes 5,6, & defueled.

c) The CVI function is required in Modes 1-6

54 **05-026** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

Then:

- A steam leak is occurring inside containment
- Containment pressure is slowly rising

Based on the above condition, which of the following completes both statements in accordance with TS 3.6.1.4, Containment Systems - Internal Pressure?

1. If containment pressure exceeds a MINIMUM of (+) __(1)__ psig, then containment DESIGN LIMITS could be exceeded during a Design Based Accident.

2. Containment internal pressure must be restored to within the limits within __(2)__.

A. (1) 1.5
(2) 1 hour

B. (1) 1.5
(2) 15 minutes

C. (1) 0.3
(2) 15 minutes

D. (1) 0.3
(2) 1 hour

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may confuse this value with the TS value of (-) 1.5 psig.

B. Incorrect. Plausible because the operator may confuse this value with the TS value of (-) 1.5 psig.

C. Incorrect. Plausible because there are other TS LCO with less than one hour limits.

D. Correct. Based on TS bases for TS 3.6.1.4, Internal Pressure must be maintained between -1.5 and +0.3 psig during Modes 1-4. Limitation on containment internal pressure ensure that (1) the containment structure Is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.5 psig, and (2) the containment peak pressure does not exceed the design pressure of 47 psig during the limiting pipe break condition. The pipe breaks considered are LOCA and steam line break. Containment Pressure must be restored within limits within 1 hour.

Technical References:	TS 3.6.1.4 and Bases
Proposed References to be provided:	None
Learning Objective:	NOS05TECHSPEC-14, Objective 12. Describe the general component and parameter categories that are addressed by Technical Specification Sections 3/4.1 through 3/4.12
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.7 / 45.7
Level:	RO
Category/System:	05-Containment System
K/A #:	103 K6.15 Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Containment System: MRSS
Importance:	3.1
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the applicant must have knowledge of how a steam leak from the Main Steam system will affect the TS assumption and limits of containment internal pressure.

55 **01-018** **Points: 1.00**

Given:

- Salem Unit 2 is performing a load reduction from 100% power
- Rod Control is in AUTO

Then:

- Console Alarm, URGENT FAILURE, is Alarming
- Rod motion has stopped

Which of the following completes both statements concerning the effect on the rod control system?

1. Rods __(1)__ be inserted in MANUAL.
2. The rods are currently held in position by energizing the __(2)__ gripper coils.

A. (1) can
(2) stationary

B. (1) can
(2) moveable

C. (1) can NOT
(2) moveable

D. (1) can NOT
(2) stationary

Answer: D

Explanation / Justification:

A. Incorrect. Part 1 is incorrect. Plausible because the operator could believe that even with an Urgent Failure present, the operator will always be able to insert rods. Incorrect in that all rod motion is inhibited in Manual and Automatic. Part 2 is correct.

B. Incorrect. Part 1 is incorrect. Plausible because the operator could believe that even with an Urgent Failure present, the operator will always be able to insert rods. Incorrect in that all rod motion is inhibited in Manual and Automatic. Part 2 is incorrect. Plausible because the moveable gripper coil is used to hold the rods in position until the lift coil is energized/de-energized to withdraw or insert rods. Incorrect in that the stationary coil is energized at a reduced current to hold rods in position.

C. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because the moveable gripper coil is used to hold the rods in position until the lift coil is energized/de-energized to withdraw or insert rods. Incorrect in that the stationary coil is energized at a reduced current to hold rods in position.

D. Correct. Per console ARP S2.OP-AR.ZZ-0012, Urgent Failure alarm will prevent all rod motion in Manual or Automatic. In this condition the rods are held in position by the stationary gripper coil being energized at a reduced current. The moveable gripper coil is de-energized at this time.

Technical References:	NOS05RODS00-14, S2.OP-AR.ZZ-0012 (R40)
Proposed References to be provided:	N/A
Learning Objective:	NOS05RODS00-14, Objective 6.b. Describe the function of the following components and how their normal and abnormal operations affects the Rod Control and Position Indication Systems: Control Rod Drive Mechanism (CRDM)
Question Source:	Modified Bank – Diablo Canyon 2019 NRC RO29
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.2 to 41.9 / 45.7 to 45.8
Level:	RO
Category/System:	01-Control Rod Drive System
K/A #:	001 K1.03 Knowledge of the physical connections and/or cause and effect relationships between the Control Rod Drive System and the following systems: CRDM
Importance:	3.7
Tier/Group	Emergency and Abnormal Plant Evolutions -Tier 2/Group 2

K/A Match: The K/A is matched because the question is asking the operator knowledge of how the CRDM is effected when an Urgent Failure alarm is present and which gripper coil is used to hold the rods in position.

Modified Question: Part 2 of the parent question was modified to ask which gripper coli is energized to hold the rods in-place.

Parent Question: Diablo Canyon 2019 NRC RO29

Question 29

Unit 2 is ramping power down from 100% power.

PK03-17, Rod Cont Urgent Failure, alarms.

1) Rods _____ be inserted in MANUAL.
2) The lift coil will be _____.

A. 1) can
   2) energized at reduced current

B. 1) can
   2) deenergized

C. 1) can NOT
   2) energized at reduced current

D. 1) can NOT
   2) deenergized

Proposed Answer: D. 1) can NOT 2) deenergized

Additional Reference: NOS05RODS00 Rod Control Lesson Plan

e) Coil Current Levels

(1) 3 Levels of current supplied to coils

(2) HIGH–Applied when coil is initially energized to ensure positive latching, then reduced

- Lift coil–40 amps
- Movable/stationary. Gripper coil–8 amps

(3) LOW–Applied during remainder of stepping sequence. Also applied to stationary coils to hold rods

- Lift coil–16 amps
- Movable gripper coil is not reduced
- Station Gripper Coil–4.4 amps

(4) OFF–coil deenergized

56 **01-019** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- Control Bank D rods are at 200 steps

Then:

- The Group Demand Step Counter on 2CC2 console for Control Bank D Group 1 rods fails (blank screen)

In accordance with S2.OP-DL.ZZ-0003, Control Room Logs, which of the following completes both statements?

1. The Plant Computer displays the rod demand step position for Group(s) __(1)__.
2. Based on the above condition, the Plant Computer __(2)__ be used to satisfy the Tech Spec requirements for the failed console group demand step counter.

A. (1) 1 Only
(2) can

B. (1) 1 and 2
(2) can NOT

C. (1) 1 and 2
(2) can

D. (1) 1 Only
(2) can NOT

Answer: A

Explanation / Justification:

A. Correct. . Only the Group 1 Demand Steps input into the Plant Computer. Per S2.OP-DL.ZZ-0003 for Control Rod Counter channel checks, either the Group Demand Counter or Plant Computer will satisfy the surveillance for Group 1 only. Only the Group 1 Demand Steps input into the Plant Computer. TS 3.1.3.2.1.b, bases states in part, "…either the control console group 1 demand step counter or the plant computer "bank step" display is sufficient to comply with this specification for group 1 rod position."

B. Incorrect. Part 1 is incorrect. Plausible because the operator may believe that both groups are also displayed on the plant computer. Part 2 is incorrect. Plausible because the operator may believe that the Plant Computer cannot satisfy the surveillance requirement. Incorrect in that per S2.OP-DL.ZZ-0003, the Plant Computer Group Demand for Group 1 only will satisfy the surveillance and also that only group 1 demand is displayed on the plant computer.

C. Incorrect. Part 1 is incorrect. Plausible because the operator may believe that both groups are also displayed on the plant computer. Part 2 is correct.

D. Incorrect. Part 1 is incorrect. Plausible because the operator may believe that the failed step counter will generate E-24. Incorrect in that E-24 does not use input directly from the console step counters. Part 2 is incorrect. Plausible because the operator may believe that the Plant Computer cannot satisfy the surveillance requirement. Incorrect in that per S2.OP-DL.ZZ-0003, the Plant Computer Group Demand for Group 1 only will satisfy the surveillance.

Technical References:	S2.OP-DL.ZZ-0003 (R126), TS 3.1.3.2.1 bases
Proposed References to be provided:	None
Learning Objective:	NOS05RODS00-14, Objective 6, Describe the function of the following components and how their normal and abnormal operations affects the Rod Control and Position Indication Systems: Rod Insertion Limit Comparator
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.6
Level:	RO
Category/System:	01-Position Indication System
K/A #:	014 A2.11 Ability to (a) predict the impacts of the following on the Position Indication System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Failed group demand position indication
Importance:	3.0
Tier/Group	Emergency and Abnormal Plant Evolutions -Tier 2/Group 2

K/A Match: The K/A is matched because the candidate is being asked to predict how the console step counter will effect rod position system and determine if the channel check surveillance can be satisfied using the plant computer.

Additional References: S2.OP-DL.ZZ-0003 Control Room Logs Modes 1-4

ATTACHMENT 1 S2.OP-DL.ZZ-0003(Q)

Page 29 of 51

MONDAY DATE: ___/___/___

CONTROL ROOM LOG (MODES 1-4)

DESCRIPTION	SURV. RQMT	INSTR NO.	UNITS	MIN	MAX	MON 1830	MON 0630	TUE 1830	TUE 0630	WED 1830	WED 0630	THU 1830	THU 0630	FRI 1830	FRI 0630	SAT 1830	SAT 0630	SUN 1830	SUN 0630
SHUTDOWN ROD COUNTERS																			
ALL ROD DEMAND COUNTERS/IRPI STEPS PER PLANT COMP	4.1.3.4.b	---	STEPS	UNSAT	SAT														
* IF ALL Plant Computer IRPI indicate ±12 steps (for >85% RTP) OR ±18 steps (for <=85% RTP) of their respective group demand counters, THEN RECORD "SAT". IRPI indication of <0 steps or >235 steps are considered 0 or 235 steps respectively.																			
CONTROL ROD COUNTERS																			
GROUP NOTE - S/R 4.1.3.1.1, 4.1.3.2.1.1 & 4.1.3.5 - Maintain Control Bank Overlap of 101 Steps. IF ALL rods in the indicated bank are within ±12 steps (for >85% RTP) OR ±18 steps (for <=85% RTP) of the group step counters, THEN RECORD "SAT". IF Reactor Trip Breakers are open, THEN RECORD "STBY". IF in Mode 1 or 2 AND EITHER the Rod Position Deviation Alarm (OHA-E24) OR the Rod Insertion Limit Monitor is inoperable, THEN RECORD Step Counter and Stepcounter/IRPI comparison every 4 hours on Attachment 4. GROUP NOTE - FOR Group Demand Counter: Either the Group Counter or the Plant Computer will satisfy the surveillance for Grp 1 only. FOR IRPI INDICATION: Either the Dixon or the Plant Computer will satisfy the surveillance. Both are recorded to provide flexibility if an indication fails. IRPI indication of <0 steps or >235 steps are considered 0 or 235 steps respectively. IF an IRPI (Dixon or Plant Computer) is UNSAT, THEN RECORD in the comments, the specific rod(s) which are affected, THEN GO TO S2.OP-AB.ROD-0001(Q).																		
BANK A DEMAND	4.1.1.1.b	---	STEPS	UNSAT	SAT														

57 **07-027** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

Then:

- The ventilation system that supplies cool air to the room that the Reference Junction Panel is located for the In-Core Temperature Monitoring System is lost
- Room temperature have risen by 15 °F

Which of the following describes the impact on the In-Core Monitoring instrumentation and what is its design limit?

A. CET readings will indicate higher than actual; up to 3300 °F.

B. CET readings will indicate higher than actual; up to 2300 °F.

C. NO effect on CET readings as they are temperature compensated; up to 2300 °F.

D. NO effect on CET readings as they are temperature compensated; up to 3300 °F.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because higher temperatures in the room where the reference junction box is located will cause the in temperature indicated to rise. Incorrect in that the system has a temperature compensator circuit such that the reference junction box temperature changes are adjusted to the true thermocouple reading. In-core system has a design range to 2300 °F.

B. Incorrect. . Plausible because higher temperatures in the room where the reference junction box is located will cause the in temperature indicated to rise. Incorrect in that the system has a temperature compensator circuit such that the reference junction box temperature changes are adjusted to the true thermocouple reading. In-core system has a design range to 2300 °F.

C. Correct. The reference junction box is located outside of containment. Ambient temperature changes in the room will affect the reference junction box, however the system uses a temperature compensator to adjust for the changes in temperatures where the reference junction box is located. A rise in ambient room temperature will raise temperature, but the temperature compensating circuit will adjust for these reference junction box temperature changes. In-Core system design upper range is 2300 °F.

D. Incorrect. In-Core system design upper range is 2300 °F.

Technical References:	INPO Sensors and Detectors (R3)
Proposed References to be provided:	None
Learning Objective:	NOS05INCORE-09, Objective 7. Describe how the following components affect the Core Exit Thermocouple System during Normal and Abnormal conditions: Reference Junction Boxes
Question Source:	Bank – Indian Point 2 2021 NRC RO58
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 / 45.7
Level:	RO
Category/System:	07-In Core Temperature Monitoring System
K/A #:	017 K6.01 Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the In Core Temperature Monitoring System: Temperature measuring device (for example thermocouple)
Importance:	3.2
Tier/Group	Emergency and Abnormal Plant Evolutions -Tier 2/Group 2

K/A Match: The K/A is matched because the candidate is being asked how the In Core Monitoring System is impacted based on ambient temperatures changes at the reference junction box.

Additional References: INPO Sensors and Detectors

Thermocouple Operation Example

Thermocouples will cause an electric current to flow in the attached circuit when subjected to changes in temperature. The amount of current produced depends on the temperature difference between the measurement and reference junction, the characteristics of the two metals used, and the characteristics of the attached circuit. The figure below shows a basic thermocouple circuit.

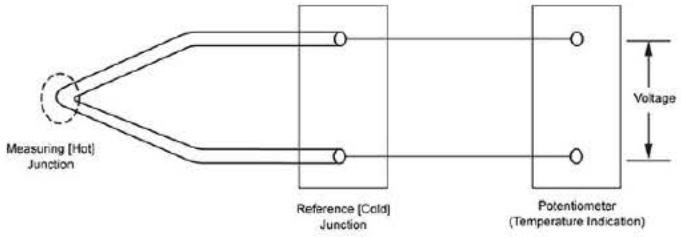


Figure: Simple Thermocouple Circuit

Heating the measuring junction of the thermocouple produces a voltage that is greater than the voltage across the reference junction. The voltmeter measures the difference between the two voltages (in millivolts); the voltage is proportional to the difference in temperature.

$$Temperature\ Indicated \propto Hot\ Junction\ Voltage - Cold\ Junction\ Voltage$$

If temperature at the reference (cold) junction were to decrease, indicated temperature would increase and vice versa. For ease of operator use, some voltmeters are set up to read out directly in temperature through use of electronic circuitry. Other applications provide only the millivolt readout. In order to convert the millivolt reading to its corresponding temperature, the operator must refer to vendor-supplied thermocouple tables. The thermocouple manufacturer supplies these tables, and they list the specific temperature corresponding to a series of millivolt readings.

Additional Reference: NOS05INCORE-06

LESSON NAME: INCORE NUCLEAR INSTRUMENTATION SYSTEM, INCLUDING THERMOCOUPLES AND SATURATION MARGIN MONITOR
NOS05INCORE-06 04/09/19

INSTRUCTIONAL CONTENT:

2) To measure the cold junction temperature with a semiconductor temperature probe, and provide that temperature signal to the CETPS so that the temperature error introduced by the cold junction may be compensated.

NOTE: The cold junction temperature is the temperature at the point where the thermocouple wires are connected to copper wires at the terminal strip. This is different than the temperature sensed by the thermocouple, because, in essence, a new thermocouple is formed by the junction between the thermocouple wires and the copper wire. This "new" thermocouple produces a voltage proportional to its junction temperature, which adds to the voltage already present from the FTTC. By measuring this temperature, the CETPS can compensate for this error to produce a true core temperature.

58 **05-027** **Points: 1.00**

Which of the following completes both statements concerning operation of the Hydrogen Recombiner System during a Design Bases Accident (assume hydrogen concentration is within the range that allows operation of the hydrogen recombiner system)?

1. The maximum number of hydrogen recombiners that can be placed in service is __(1)__.
2. What is the concern if hydrogen recombiner system cannot be placed in service when required?

A. (1) 1
(2) Gas vapor binding of ECCS pumps resulting in loss of core cooling.

B. (1) 1
(2) Explosive atmosphere resulting in containment failure.

C. (1) 2
(2) Explosive atmosphere resulting in containment failure.

D. (1) 2
(2) Gas vapor binding of ECCS pumps resulting in loss of core cooling.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because gas intrusion into the ECCS system can adversely affect the operation of ECCS pumps.

B. Correct. Per 2-EOP-LOCA-1 step 17, only one Hydrogen Recombiner is allowed to be placed in service when the Hydrogen concentration is between > 0.5 % and < 4%. The bases for running a Hydrogen Recombiner is to reduce the hydrogen concentration to prevent the formation of a flammable mixture that could ignite and cause an explosion and pressure excursion that could challenge containment integrity.

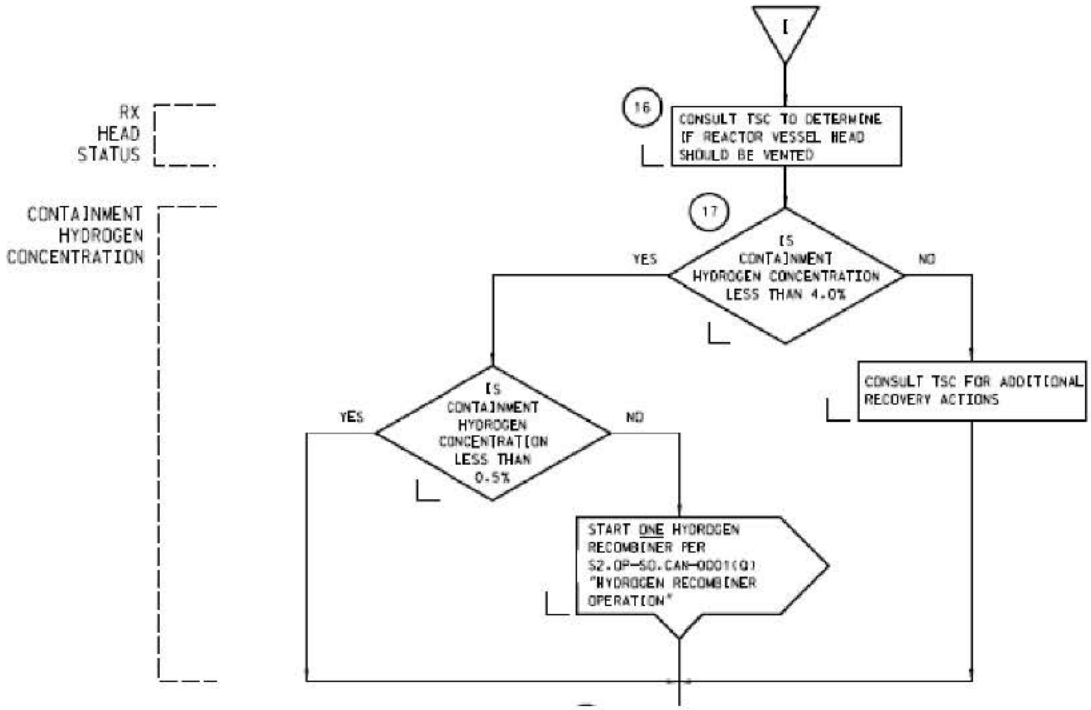
C. Incorrect. Plausible because the operator may believe that both hydrogen combiners can be operated to maximize the reduction of hydrogen when hydrogen concentration is at the upper end (4%). Incorrect in that operation of both hydrogen recombiners is not permitted.

D. Incorrect. Plausible because gas intrusion into the ECCS system can adversely affect the operation of ECCS pumps. Plausible because the operator may believe that both hydrogen combiners can be operated to maximize the reduction of hydrogen when hydrogen concentration is at the upper end (4%). Incorrect in that operation of both hydrogen recombiners is not permitted.

Technical References:	2-EOP-LOCA-1 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05LOCA01-07, Objective 6.j. Describe the plant response to actions taken in the following EOP step sequence(s): Containment Hydrogen Concentration
Question Source:	Bank - Salem
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 / 45.4
Level:	RO
Category/System:	05-Hydrogen Recombiner and Purge Control System
K/A #:	028 K3.02 Knowledge of the effect that a loss or malfunction of the Hydrogen Recombiner and Purge Control System will have on the following systems or system parameters: CNT
Importance:	2.9
Tier/Group	Plant Systems-Tier 2/Group 2

K/A Match: The K/A is matched because the question is asking the applicant what the effect on containment will be when hydrogen recombiners cannot be placed in service when hydrogen concentration requires it in EOP-LOCA-1.

Additional Reference: 2-EOP-LOCA-1



Additional Reference:

ERG Step No: Step 17

EOP Step:

IS CONTAINMENT HYDROGEN CONCENTRATION LESS THAN 4.0%
[CONTAINMENT HYDROGEN CONCENTRATION]

Purpose:

To check if an excessive containment hydrogen concentration is present.

ERG Basis:

This step instructs the operator to obtain a current hydrogen concentration measurement. Depending upon the magnitude of the hydrogen concentration, the operator will either continue with EOP-LOCA-1, turn on the hydrogen recombiners or notify the plant engineering staff to determine additional recovery actions before continuing with the procedure.

When inadequate core cooling has occurred, the containment hydrogen concentration may be as much as 10 to 12 volume percent, depending on the amount of metal-water reaction (to produce hydrogen) that has occurred in the core. The hydrogen concentration is of concern since a flammable mixture can burn, if an ignition source is available, and cause a sudden rise in containment pressure which may challenge containment integrity. The operator is instructed to obtain a current containment hydrogen concentration measurement at this point in order to ascertain the potential flammability of the combustible gases in the containment. Note that in order to have the potential for flammable hydrogen concentrations, an inadequate core cooling situation must have already existed. Without an inadequate core cooling situation, sufficient hydrogen would not be expected to have been produced to cause potentially flammable mixtures.

A determination is made of the flammability of the hydrogen mixture with respect to the possible containment pressure rise. If the hydrogen mixture is between 0.5 volume percent and 6.0 volume percent in dry air, either no hydrogen burn is possible or a limited burn may occur which does not produce a significant pressure rise. A hydrogen concentration not to exceed 6.0 volume percent in dry air corresponds to the upper limit of operability for the hydrogen recombiner, represented by the footnote (T.18). If containment hydrogen concentration is between 0.5 volume percent and(T.18) volume percent, the operator is instructed to start the Hydrogen Recombiner System to slowly reduce containment hydrogen concentration. If the hydrogen concentration is less than 0.5 volume percent in dry air, a flammable situation is not imminent and the operator continues with procedure EOP-LOCA-1. If the concentration is greater than (T.18) volume percent in dry air, the operator is instructed to immediately notify the plant engineering staff of the situation. In this case the operator is instructed to consult the plant engineering staff for additional recovery actions while proceeding with this procedure.

59 **08-023** **Points: 1.00**

Given:

- Salem Unit 1 is in MODE 5
- Spent Fuel Pool level is 128' 5"
- Spent Fuel Pool boron concentration is 2300 ppm

Then:

- A tube inside the in-service Spent Fuel Pool Heat Exchanger ruptures

Which of the following completes both statements?

1. Level in the Spent Fuel Pool will __(1)__.
2. The Spent Fuel Pool boron concentration will require adjustments from a __(2)__ water source.

A. (1) rise
(2) non-borated

B. (1) rise
(2) borated

C. (1) lower
(2) borated

D. (1) lower
(2) non-borated

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the SFP level will rise. Incorrect in that CCW is a non-borated water source and will reduce the boron concentration.

B. Correct. Per S2.OP-AB.CC-0001, a leak in the SFP HX will result in the out leakage of CCW into the SFP system resulting in the CCW Surge Tank level lowering and the SFP level rising. The non-borated CCW water will reduce the boron concentration of the SFP. If boron concentration lowers below TS then the crew will need to add borated water into the SFP to raise the boron concentration. Note: SFP low and high level alarms 128' 2" and 129' 2".

C. Incorrect. Plausible because the operator may incorrectly recall that the SFP HX will leak into the CCW system. Incorrect per S1.OP-AB.CC-0001, SFP HX will be out leakage from CCW.

D. Incorrect. Plausible because the operator may incorrectly recall that the SFP HX will leak into the CCW system. Incorrect per S1.OP-AB.CC-0001, SFP HX will be out leakage from CCW and that the boron concentration will reduce requiring addition from a borated source.

Technical References:	S1.OP-IO.ZZ-0010 (R30), TS 3.9.11
Proposed References to be provided:	None
Learning Objective:	NOS05SFP000-12, Objective 5.e. Describe the function of the following components and how their abnormal operation affects the Spent Fuel Pool Cooling System: Spent Fuel Pool Heat Exchanger
Question Source:	Bank – Turkey Point 2016 NRC RO61
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 / 45.7
Level:	RO
Category/System:	08-Spent Fuel Pool Cooling System
K/A #:	033 K6.11 Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Spent Fuel Pool Cooling System: CCWS
Importance:	3.4
Tier/Group	Plant Systems-Tier 2/Group 2

K/A Match: The K/A is matched because the question is asking the applicant what the effect will be on the SFP when a tube leak exists in the SFP heat exchanger. In this case, CCW will leak into the SFP system.

Additional References: Turkey Point 2016 RO61

Question # 61

Given the following initial conditions:

- Unit 3 is in Mode 5.
- Spent Fuel Pit (SFP) level is 56' 11".
- SFP boron concentration is 2300 ppm.

Subsequently:

- A tube inside of the in-service Spent Fuel Pool Heat Exchanger breaks.
- ANN H 8/6 CCW HEAD TANK HI/LO LEVEL alarms.
- A crew is dispatched to isolate the heat exchanger.

Which one of the following correctly completes the statement below?

The crew must take action to prevent Spent Fuel Pool ___(1)___ from violating the Tech Spec LCO by performing a ___(2)___ to the SFP in accordance with 3-NOP-033, Spent Fuel Pit Cooling System.

A. (1) level
(2) direct boration

B. (1) level
(2) primary water fill

C. (1) boron concentration
(2) direct boration

D. (1) boron concentration
(2) primary water fill

Additional Reference: S1.OP-IO.ZZ-0010

ATTACHMENT 1
(Page 5 of 5)
SUPPORT SYSTEMS STATUS

3.0 SYSTEMS/COMPONENTS REQUIRED OPERABLE:

System/Component (14)		Actual Status	Notes	Comments
FHB VENTILATION	(10)			
FUEL HANDLING CRANES			TRM 3.9.7	
SFP BORON CONCENTRATION	(11)	Cb: ____ Date: ____	Cb Required: T/S 3.7.11, T/S 3.9.1, FSAR 9.1.3	
CONTROL ROOM EMERGENCY AIR CONDITIONING SYSTEM	(12)		T/S 3.7.6.1	
1R1B RAD MONITORING	(13)		T/S 3.3.3.1	
CHILLED WATER SYSTEM			T/S 3.7.10	

(10) The FHB Supply Fan AND both FHB Exhaust Fans should be AVAILABLE AND operating. Each FHB Exhaust Fan is to be aligned to an available Vital Bus. A FHB Exhaust Fan may be aligned to 1C Vital Bus as an alternate power source IAW S1.MD-FR.ZZ-0001(Q), Alternate Power Sources During Refueling Outages. IF Fuel Handling Ventilation becomes unavailable, operations may continue after the Fuel Handling Area ventilation system has been evaluated and building D/P verified to be negative.

(11) The SFP Boron concentration (Cb) shall be greater than or equal to800 ppm. Applicable when fuel assemblies are stored in the SFP and a fuel storage pool verification has NOT been performed since the last movement of fuel assemblies in the SFP (T/S 3.7.11).
In Mode 6, the SFP Boron concentration (Cb) shall be within the limits specified in the Core Operating Limits Report (COLR). Applicable when the SFP is connected to the Reactor Coolant System (T/S 3.9.1).
For Dry Cask Storage Campaigns, SFP required Boron concentration (Cb) shall be provided by Reactor Engineering IAW SC.RE-FR.DCS-0002(Q).
The SFP Boron concentration (Cb) is maintained at a minimum of 800 ppm (FSAR 9.1.3).

(12) For fuel movement, the CREACS fans do NOT require an Emergency Backup power supply to be considered OPERABLE.

(13) Radiation monitors are inoperable with the system in ACCIDENT PRESSURIZED mode. This is an acceptable condition.

(14) **REFER** to S1.OP-SO.DG-0005(Q), Preparation For Removing A Diesel Generator From Service, for "Required Minimum Components Necessary For Movement Of Irradiated Fuel", based on Emergency Diesel Generator status.

Additional Reference: S1.OP-AB.CC-0001

___ 3.35 **CONTINUE** efforts to locate and isolate the leak, as applicable.

___ 3.36 **EVALUATE** initiating Unit shutdown IAW the applicable IOP.

NOTE

___ Components supplied by CCW with pressure less than CCW HDR pressure are sources for system to system out leakage (e.g., Spent Fuel Pool Heat Exchanger).

___ 3.37 **ISOLATE** the affected Component using ATTACHMENT 3, Equipment Isolation Table.

3.38 **WHEN** source of component leakage has been identified and isolated,

___ **THEN ENSURE** any valves that were CLOSED to support leak detection IAW ATTACHMENT 4, Leak Detection Method, are realigned to support normal Component Cooling System operation.

___ 3.39 **NOTIFY** SM/CRS to refer to the following:

___ ♦ Technical Specifications

___ ♦ Event Classification Guide

60 **04-079** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

Then:

- A tube in one of the Circulating Water (CW) waterboxes completely severs

Which of the following control room console alarms would confirm a gross tube leak in a CW waterbox?

1) Dissolved Oxygen Hi
2) Hydrazine Lo
3) Condensate Pump Disch Sodium Hi
4) Hotwell Outlet Conductivity Hi

A. 1 and 3 Only

B. 2 and 4 Only

C. 1, 3, and 4 Only

D. 2, 3, and 4 Only

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because these two alarms would confirm a waterbox tube leak. Incorrect in that the answer is incomplete and missing Hotwell Outlet Conductivity Hi.

B. Incorrect. Plausible because Dissolve Oxygen alarm would be expected and Hydrazine Lo alarm is an alarm for the feedwater system. Incorrect in that it's not expected for river water into the condensate system to impact the concentration of Hydrazine.

C. Correct. Based on a severed tube in a waterbox, these alarms would confirm a tube leak in a CW waterbox.

D. Incorrect. Plausible because these alarms would confirm a tube leak. Incorrect in that it's not expected for river water into the condensate system to impact the concentration of Hydrazine.

Technical References:	S2.OP-AB.CHEM-0001 (R30)
Proposed References to be provided:	None
Learning Objective:	NOS05CONTMT-17, Objective 3.g,
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.3 / 45.12
Level:	RO
Category/System:	04-Condensate System / Emergency Procedures/Plan
K/A #:	056 G2.4.46 Ability to verify that the alarms are consistent with the plant conditions
Importance:	4.2
Tier/Group	Plant Systems-Tier 2/Group 2

K/A Match: The K/A is matched because the question is asking the applicant to verify which alarms in the control room would be present on the Condensate System if a CW waterbox tube leak exists.

Additional Reference:

S2.OP-AB.CHEM-0001(Q)

ATTACHMENT 2
CONDENSER TUBE LEAK
(Page 1 of 5)

1.0 **Actions for Condenser Tube Leak**

NOTE

___ A gross Condenser Tube Failure/Leak is indicated by rapidly rising Secondary Plant Chemistry recorder trends, 2CC1 and OHA alarms, it is also defined as:

___ ≥30 ppb Condensate Pump Disch sodium with condensate polisher in service, OR

___ ≥2 ppb Condensate Pump Disch sodium with condensate polisher out of service, OR

___ A rapidly increasing Condensate Pump discharge sodium in either condition (condensate polishers in or out of service) that is expected to exceed the limits above (>30 or >2 ppb).

2RP5 Recorders

___ Conductivity 2KA-8551
___ Sodium, Specific CNDCT

2CC1 Alarms

___ CONDENSATE PUMP DISCH SODIUM HI
___ HOTWELL OUTLET CONDUCTIVITY HI
___ CONDENSATE PUMP DISCH CONDUCTIVITY HI

OHA Alarms

___ COND POL AUTO SWAP (G-19)

61 **09-007** **Points: 1.00**

Given:

- Salem Unit 1 is at 100% power and Unit 2 is in MODE 5.
- Release of 11 CVCS Monitor Tank is in progress.
- The release path is through 11 CC Heat Exchanger and Unit 2 CW system
- 22A and 22B Circulators are the only circulators in service

Then:

- 22A Circulator trips

Based on the above condition, which one of the following describes the impact to the liquid release in progress?

1) The release __(1)__ automatically isolate upon the loss of the circulator.
2) The release __(2)__.

A. (1) will NOT
(2) may continue

B. (1) will NOT
(2) may NOT continue

C. (1) will
(2) may NOT recommence

D. (1) will
(2) may recommence

Answer: A

Explanation / Justification:

A. Correct. Per S1.OP-SO.WI-0001 only one circulator is required to be in service to allow the release to continue. The 1R18 does not isolate or close the 1WL51 on loss of flow.

B. Incorrect. Plausible because the operator may believe that you need both circulators in service to continue the discharge.

C. Incorrect. Plausible because the operator may believe that the 1R18 radiation monitor will trip the 1WL51 closed when loss of flow is detected or may believe the operator may believe that you need both circulators in service to continue the discharge.

D. Incorrect. Plausible because the operator may believe that the 1R18 radiation monitor will trip the 1WL51 closed when loss of flow is detected.

Technical References:	S1.OP-SO.WL-0001 (R29)
Proposed References to be provided:	None
Learning Objective:	NOS05WASLIQ-11, Objective 12. Discuss the procedural requirements associated with the Radioactive Liquid Waste System, including an explanation of major precaution and limitations in the Radioactive Liquid Waste System procedures.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.3
Level:	RO
Category/System:	08-Liquid Radwaste System
K/A #:	068 K5.07 Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Liquid Radwaste System: Loss of secondary mixing water system
Importance:	2.6
Tier/Group	Plant Systems-Tier 2/Group 2

K/A Match: The K/A is matched because the applicant is being asked how the loss of one circulator will affect the liquid release in progress.

62 **07-028** **Points: 1.00**

Given:

- Salem Units 1 and 2 are at 100% power

Then:

- 1R1A, Control Room Area, Radiation Monitor is in Alarm

Which of the following actions is required?

Note: CAV = Control Room Area Ventilation
AP = Accident Pressurized

A. Verify automatic actuation of ONLY Unit 1 CAV in AP Mode.

B. Verify automatic actuation of ONLY Unit 2 CAV in AP Mode.

C. Manually press AP pushbutton at 1RP2 panel to actuate BOTH units CAV in AP Mode.

D. Manually press AP pushbutton at 2RP2 panel to actuate BOTH units CAV in AP Mode.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because the operator may confuse the 1R1A with the 1R1B radiation monitors and believe that it will actuate CAV in AP Mode only on the affected unit. Incorrect in that 1R1A area radiation monitors will not actuate CAV in AP Mode.

B. Incorrect. Plausible because the operator may confuse the 1R1A with the 1R1B radiation monitors and believe that it will actuate CAV in AP Mode only on the affected unit. Incorrect in that 1R1A area radiation monitors will not actuate CAV in AP Mode.

C. Correct. Per the ARP for 1R1A radiation monitor in alarm, the control room crew will be required to place the Control Room Ventilation (CAV) System into Accident Pressurized (AP) Mode. Since the area radiation monitor is alarm sensing radiation on Salem Unit 1 side, the crew will actuate CAV in AP Mode from Unit 1 which will actuate both units CAV into AP Mode and open the emergency intake dampers on the unaffected unit, in this case unit 2.

D. Incorrect. Plausible because the operator may confuse which unit to actuate CAV in AP Mode to align the emergency intake dampers on the unaffected unit, in this case unit 2.

Technical References:	S2.OP-AB.RAD-0001 (R30), S2.OP-SO.CAV-0001 (R43)
Proposed References to be provided:	None
Learning Objective:	NOS05REFUEL-13,
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 / 45.5 to 45.8
Level:	RO
Category/System:	07-Area Radiation Monitoring System
K/A #:	072 A4.02 Ability to manually operate and/or monitor the Area Radiation Monitoring System in the control room: Radiation monitor function
Importance:	3.4
Tier/Group	Plant Systems-Tier 2/Group 2

K/A Match: The K/A is matched because the question is asking what action is required when the control room area radiation monitor goes into alarm.

Additional Reference:

S1.OP-AB.RAD-0001(Q)

ATTACHMENT 3
(Page 1 of 1)

AREA RADIATION MONITORS

___ 1.0 IF 1R1A, Control Room Area, is the affected monitor, THEN:

___ 1.1 Coordinate with Unit 2 NCO to **ALIGN** Control Room Ventilation for ACCIDENT PRESSURIZED mode IAW the following procedures:

- S1.OP-SO.CAV-0001(Q), Control Room Ventilation Operation
- S2.OP-SO.CAV-0001(Q), Control Room Ventilation Operation

___ 1.2 DIRECT Radiation Protection to survey area to determine source of activity rise.

Additional Reference:

5.7 Manual Initiation of ACCIDENT PRESSURIZED Mode Operation

NOTE

___ This Section provides direction for placing both Units in the ACCIDENT PRESSURIZED mode manually initiated by Unit 1. If ACCIDENT PRESSURIZED mode was initiated from Unit 2, Step 5.7.1 should NOT to be performed.

___ 5.7.1 **PRESS** ACCIDENT PRESSURIZED pushbutton at 1RP2.

NOTE

___ IF ACCIDENT PRESSURIZED mode was initiated from Unit 2,
THEN NORMAL lamp will remain illuminated.

___ 5.7.2 **ENSURE** red ACCIDENT PRESSURIZED lamp illuminates.

___ 5.7.3 **ENSURE** CONTROL AREA ISOLATION Train A ISOLATION PUSH TO RESET red lamp is illuminated AND Train B ISOLATION PUSH TO RESET red lamp is illuminated.

___ 5.7.4 IF an accident has occurred on Unit 1,
THEN **NOTIFY** Unit 2 NCO to PERFORM S2.OP-SO.CAV-0001(Q), Control Area Ventilation Operation, section on Automatic Initiation of ACCIDENT PRESSURIZED Mode Operation.

___ 5.7.5 IF manual initiation of ACCIDENT PRESSURIZED mode operation is NOT initiated due to an accident,
THEN **NOTIFY** Unit 2 NCO to PERFORM the following:

___ A. **ENSURE** Unit 2 CAACS is aligned to ACCIDENT PRESSURIZED mode of operation IAW S2.OP-SO.CAV-0001(Q), Control Area Ventilation Operation, Exhibit 1 for Component Positions.

63 **08-024** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- OHA A-7, FIRE PROT FIRE, is alarming
- Fire alarm zones on 2RP5 Panel indicates a fire detected inside Panel 335 in containment

Which of the following completes the following statement concerning the operation of the 2FP147, Fire Protection Containment Isolation Valve?

1. Based on a valid fire detected inside containment, the 2FP147 will __(1)__.
2. On a Phase A signal, the 2FP147 __(2)__ automatically close.

A. (1) automatically open
(2) will NOT

B. (1) automatically open
(2) will

C. (1) need to be manually opened from the 2RP5 panel
(2) will

D. (1) need to be manually opened from the 2RP5 panel
(2) will NOT

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because fire protection systems will actuate if the fire detector sensor is triggered. Incorrect in that even though the fire zone fire water header is triggered, the 2FP147 still needs to be manually opened from the control room.

B. Incorrect. Plausible because fire protection systems will actuate if the fire detector sensor is triggered. Incorrect in that even though the fire zone fire water header is triggered, the 2FP147 still needs to be manually opened from the control room.

C. Correct. The 2FP147 requires to be manually opened at the 2RP5 panel when a valid fire is detected inside containment. The 2FP147 will close on a Phase A signal.

D. Incorrect. Plausible because the operator may believe that fire water inside containment is the priority and will not auto isolate.

Technical References:	NOS05FIRPRO-12
Proposed References to be provided:	None
Learning Objective:	NOS05FIRPRO-12, Objective 13. Given plant conditions, relate the Fire Protection System with the following: Containment and Containment Press Relief
Question Source:	Modified – Salem 2019 NRC RO75
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.5 / 45.5
Level:	RO
Category/System:	08-Fire Protection System
K/A #:	086 A1.06 Ability to predict and/or monitor changes in parameters associated with operating the Fire Protection System, including: Lights and alarms
Importance:	3.0
Tier/Group	Plant Systems-Tier 2/Group 2

K/A Match: The K/A is matched because the question is asking the applicant how the fire water is supplied into containment when a valid fire is detected. The operator requires knowledge of hoe the 2FP147 fire header isolation valve functions during a valid fire detected and whether the 2FP147 will isolate on a Phase A containment isolation signal..

Additional Reference:

ALARM **A-7**	7 FIRE PROT FIRE
DEVICES: N/A SETPOINT: N/A REFERENCES: Dwg. 203753, 205447, 601270, 203750, 203751	

A-7 FIRE PROT FIRE

1.0 CAUSE(S):

1.1 Any:

A. Coded fire alarm

B. Fire alarm on 2RP5

2.0 AUTOMATIC ACTION:

NONE

3.0 OPERATOR ACTIONS:

3.1 **SCAN** 2RP5 to determine Fire Zone, Area, and Elevation
OR **DETERMINE** location IAW Alarm Code List.

3.2 IF AT ANY TIME, fire indication for both Zones 59 and 74 on 2RP5 are received,
THEN OPEN 2FP147, Fire Protection Containment Isolation.

Parent Question: Salem 2019 NRC RO75

Modified question based on the stem asking about isolation on a Phase A.

Question: #75

Given:

- Unit 2 is operating at 100 % Power.
- OHA A-7, FIRE PROT FIRE, annunciates.
- Panel 2RP5 is checked and indicates a Fire in Containment.
    - Zone 59 – Air and Water Deluge, Containment El. 100 Panel 335 is lit.
    - Zone 74 – Smoke and Fire Detector, Containment El. 100 Panel 335 is lit.
- The crew enters S2.OP-AB.FIRE-0001, Control Room Fire Response.
- The crew Trips the Reactor, Turbine, and all RCPs.
- 2-EOP-TRIP-1 is entered while continuing with S2.OP-AB.FIRE-0001, Control Room Fire Response.

Which ONE of the following describes the **NEXT** required action for the above conditions in accordance with S2.OP-AB.FIRE-0001?

A. OPEN the 2FP147 from the control room

B. Dispatch an NEO to OPEN the associated deluge valves.

C. Dispatch an NEO to place both PORV BLOCK Valve breaker key switches in EMER CLOSE.

D. Verify OHA A-15, FIRE PUMP 1/2 RUN, is in alarm indicating that a Diesel Fire Pump has started and is supplying fire protection water to the associated deluge valves in containment.

Answer: A

64 **14-087** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- All systems are in normal alignment and in Auto

Then:

- 21 SGFP trips

Which of the following indications is the expected response?

1. The DEHC will automatically runback the turbine at __(1)__ per minute.
2. 2CN47, 23/24/25 Heater String Bypass valve, __(2)__ immediately open when the feedpump trips.

A. (1) 15%
(2) will NOT

B. (1) 15%
(2) will

C. (1) 200%
(2) will

D. (1) 200%
(2) will NOT

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because this is the ramp rate when manually initiating a turbine runback at the DEHC panel. Plausible because on a condensate pump trip, the 2CN47 will only open if SGFP suction pressure lowers to less than 320 psig. Incorrect in that the 2CN47 will open following a SGFP trip regardless of suction pressure.

B. Incorrect. Plausible because this is the ramp rate when manually initiating a turbine runback at the DEHC panel.

C. Correct. A SGFP will automatically initiate a turbine runback at 200% per minute to 66% power. When a SGFP trips the 2CN47 will open and the polisher will be bypassed.

D. Incorrect. Plausible because on a condensate pump trip, the 2CN47 will only open if SGFP suction pressure lowers to less than 320 psig. Incorrect in that the 2CN47 will open following a SGFP trip regardless of suction pressure.

Technical References:	S2.OP-AB.CN-0001 (R32)
Proposed References to be provided:	None
Learning Objective:	NOS05ABCN01-07, Objective 1.d. Describe the operation of the following system as applied to AB.CN-0001: SGFP Trip
Question Source:	Bank - Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 45.12
Level:	RO
Category/System:	14-Conduct of Operations
K/A #:	G2.1.19 Ability to use available indications to evaluate system or component status.
Importance:	3.9
Tier/Group	General Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the candidate is being asked what the expected indications and status of components are following a SGFP trip. In this case, the candidate must be able to determine from various control room indications (i.e., CN47 opening, turbine runback status) if the system is responding as expected.

65 **14-088** **Points: 1.00**

Given:

- Salem Unit 1 experienced a valid demand for a reactor trip and Safety Injection (SI)
- The RO suspects that only one Train of SI actuated

Which of the following indications would corroborate that one train of SI failed to actuate?

The "SI & FW ISOL" Red lamp on 2RP4 panel will be __(1)__ and the "SI RESET" Green bezel light on the failed Train of Safeguards on 2CC1 console will be __(2)__.

A. (1) Flashing
(2) LIT

B. (1) Flashing
(2) Extinguished

C. (1) LIT
(2) Extinguished

D. (1) LIT
(2) LIT

Answer: A

Explanation / Justification:

A. Correct. The "SI & FW ISOL" Red lamp on 2RP4 would be flashing indicating SSPS train disagreement and that one Train of SI has failed to actuate. Also, the "SI REST" Green light on the train that failed to actuate will be LIT

B. Incorrect. Plausible because the "SI & FW ISOL" Red lamp on 2RP4 would be flashing indicating SSPS train disagreement and that one Train of SI has failed to actuate. The SI RESET light would be extinguished but for the Train that did actuate

C. Incorrect. Plausible because "SI & FW ISOL" Red lamp on 2RP4 would be LIT solid if both trains of SI did actuate and the SI RESET light would also be extinguished for the train that SI actuated on.

D. Incorrect. Plausible because "SI & FW ISOL" Red lamp on 2RP4 would be LIT solid if both trains of SI did actuate and the SI RESET light would be LIT for the train that SI did not actuated on.

Technical References:	NOS05RXPROT-15, 218489
Proposed References to be provided:	None
Learning Objective:	NOS05RXPROT-15, Objectives 19a & b. Identify and describe the Control Room controls, indications, and alarms associated with the Reactor Protection System
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 45.12
Level:	RO
Category/System:	14-Conduct of Operations
K/A #:	G2.1.31 Ability to locate control room switches, controls, and indications and to determine whether they correctly reflect the desired plant lineup.
Importance:	4.6
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the candidate is being asked to determine how to validate if one train of SI has actuated and the location of the indication. In this case, the applicant must know the indication in the stem is located on 2RP4 and should be flashing to indicate train disagreement,

Additional Reference: Excerpt from NOS05RXPROT

C. RP4
   1. Indication only
   2. Used to verify a reactor trip demand
   3. Used to verify safeguards equipment status
   4. If light flashes, Train A & B do not agree. One multiplexer switch must be in "A + B" for this to occur.
      a. This Train Disagreement feature of RP4 applies to all the bistable indicating lights, plus the blue AUTO SI BLOCK light.
      b. It does NOT apply to the blue block lights associated with the blockable reactor trips (for example, SR Hi Flux) and the blockable SI's (for example, Low PZR Pressure). For these functions, RP4 shows "Train B" as an input to the 2/2 logic function – in other words – it takes both Train A and Train B to illuminate the light, and there is no flashing feature.

D. General Warning Alarm System
   1. Warns operator if SSPS is not entirely operational
   2. Illuminates a light on logic bay panel

66 **08-025** **Points: 1.00**

Which of the following completes the statement concerning Salem Unit 1 vs. Unit 2 Stator Water Cooling System design?

Salem Unit __(1)__ is the ONLY unit that has an automatic Turbine Runback feature when the following stator water system condition(s) exist; __(2)__.

A. (1) 1
(2) low flow, low pressure, or high temperature

B. (1) 2
(2) low flow, low pressure, or high temperature

C. (1) 2
(2) low flow Only

D. (1) 1
(2) low flow Only

Answer: B

Explanation / Justification:

A. Incorrect. Part 1 is incorrect. Plausible because the operator may confuse which unit has this design feature. Part 2 is correct.

B. Correct. Salem Unit 2 is the only unit that has an automatic turbine runback when any of the following conditions exist; low flow, low pressure, and high temperature.

C. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because this condition is used on both units, but unit 2 uses it for the turbine runback whereas unit 1 uses it for a turbine trip feature only.

D. Incorrect. Part 1 is incorrect. Plausible because the operator may confuse which unit has this design feature. Part 2 is incorrect. Plausible because this condition is used on both units, but unit 2 uses it for the turbine runback whereas unit 1 uses it for a turbine trip feature only.

Technical References:	NOS05SWCU02-09
Proposed References to be provided:	None
Learning Objective:	NOS05SWCU02-09, Objective 11. Identify the differences between Unit 1 and Unit 2 Stator Cooling Water System components, parameters, and operation
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.5 / 41.6 / 41.7 / 41.10 / 45.12
Level:	RO
Category/System:	14-Equipment Control
K/A #:	G2.2.3 Knowledge of the design, procedure, and/or operational differences between units.
Importance:	3.8
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the candidate is being asked the operational design difference between the units stator water cooling system. In this case, the applicant needs to know that only unit 2 has an automatic turbine runback feature.

Additional References: NOS05SWCU02

V. UNITS 1 & 2 DIFFERENCE

A. Unit 2 is a General Electric SCW Unit. Loss of flow, low pressure and hi temperature are inputs to the turbine runback circuit. If runback fails to lower amps to the specified value within the specified time delay then a turbine (generator) trip will occur.

B. Unit 1:

1. Is a Westinghouse SCW Unit (covered in a separate lesson plan).
2. Low flow inputs into a 45 second time delayed turbine trip circuit.
3. Does not have a SLMS System.

67 **14-089** **Points: 1.00**

Given:

- A valve lineup is being performed on a system that requires Independent Verification
- The valve lineup contains a valve that is required to be throttled open two turns
- Permission has been granted to move the valve as required to verify its position

What are the actions necessary to verify the position of this throttled open valve?

A. One operator will go out ALONE and fully close the valve, then reopen it two full turns. Then, the second operator will go out ALONE and verify the valve is properly positioned by visual observation of the valve stem.

B. One operator will go out ALONE and fully open the valve, then close the valve the same number turns. Then, the second operator will go out ALONE and verify the valve is properly positioned by visual observation of the valve stem.

C. One operator will go out ACCOMPANIED by a second operator. One of the operators will fully close the valve, then reopen it two full turns while the second operator observe the actions.

D. One operator will go out ACCOMPANIED by a second operator. One of the operators will fully open the valve, then close the valve the same number of turns while the second operator observe the actions

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because typical independent verification of valve positions is performed after the first operator performs the valve manipulation and the second operator goes out alone to independently verify its position.

B. Incorrect. Plausible because typical independent verification of valve positions is performed after the first operator performs the valve manipulation and the second operator goes out alone to independently verify its position.

C. Correct. Per OP-AA-108-101-1001 and OP-AA-1008-101-1002, when a throttled valve position requires verification, then the valve is fully closed, then reopened to the required number of turns. The independent verification is performed by a second operator by either; independently counting the number of turns, verification of process variable, or verification of valve position for valves with incremental indicators against the tagout footnote or procedure.

D. Incorrect. Plausible because the operator may confuse the direction of how the throttled valve must be positioned prior to returning to its throttled position. Fully closed vs. fully open.

Technical References:	OP-AA-108-101-1001 (R2), OP-AA-101-1002 (R11)
Proposed References to be provided:	None
Learning Objective:	NOS05CONDOP-14, Objective 5. Describe requirements for the following Control Room or Field Activities in accordance with applicable Conduct of Operations Manual Administrative Procedures: Shift Relief and Turnover
Question Source:	Bank – Millstone U2 NRC RO69
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 43.3 / 45.13
Level:	RO
Category/System:	14-Equipment Control
K/A #:	G2.2.14 Knowledge of the process for controlling equipment configuration or status.
Importance:	3.9
Tier/Group	General Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the candidate is being asked how to verify a throttled open valve position and independent verification.

Additional References: OP-AA-108-101-1001

4.4 Checking Manual Throttle Valve Position

4.4.1. Valves that are required to be placed in a throttled position will normally be checked in the proper position by using the appropriate process parameter (flow, pressure, etc.).

4.4.2. When operation of a throttled valve is necessary to determine its position, **CONSULT** with the appropriate Operations supervisor prior to manipulating the throttled valve for guidance, including discussion with cognizant Engineering personnel as necessary.

NOTE Personnel must be aware of the effect that mechanical valve operator slack will have on the number of hand-wheel turns required to position a valve. Compensate as necessary to allow for the slack.

1. Provided system configuration allows valve movement, check throttled valve position by counting the number of 360-degree turns required to fully close the valve.

2. **RE-POSITION** the valve to the throttled position by opening the valve the same number of turns counted in the previous step. Restoring locked throttled valves requires concurrent verification.

Additional Reference: OP-AA-108-101-1002

ATTACHMENT 5
INDEPENDENT/CONCURRENT VERIFICATION
Page 3 of 4

2.4.3. For Hope Creek valves BC-HV-F077, BC-HV-F060A, BC-HV-F060B valves should be verified "open" by observation of the valve local position indication or by use of the valve remote position indication and verified "locked" by observation that a Locking Wire is Secured Between the Valve Yoke and the Wire Loop Installed on Valve Actuator Stem. **[DCP 80072763]**

2.5 When a manual valve is to be verified closed, move the handwheel in the close direction only. Do not exceed normal closing torque. **[CD-095B, CD-088B]**

NOTE

Valves that are binding on the backseat will give a false indication of being closed. **[CD-769F]**

Valves with Safety Tags are controlled by the tagging process and do not require verification.

2.6 When a manual valve is to be verified open move the handwheel in the closed direction enough to verify valve movement. Then return the valve to the open position. **[CD-095B, CD-088B]**

2.7 Throttle valves:

2.7.1. Initial alignment or verification:

1. Fully close the valve.
2. Opened the valve the required number of turns, or if applicable, match the alignment marks.

2.7.2. Independent verification, the second operator may observe the first operator positioning the valve, then perform either an independent: **[CD-286B]**

- Count of the turns open.
- Verification of process variable.
- Verification of valve position for valves with incremental indicators against the WCM footnote or procedure.

68 **14-090** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- Release of 21 CVCS Monitor Tank is in progress per S2.OP-SO.WL-0001, Release of Radioactive Liquid Waste from 21 CVCS Monitor Tank

Then:

- 2FR1064, Radwaste Overboard Discharge Flow Recorder, fails low

Which of the following completes both statements?

1. Based on the above condition, the 2WL51, Overboard Valve, __(1)__ automatically close..
2. Per the ODCM with 2FR1064 Flow recorder inoperable, the release __(2)__.

A. (1) will
(2) may recommence if effluent flows are estimated once per 4 hours.

B. (1) will NOT
(2) may NOT recommence unless 2FR1064 is OPERABLE

C. (1) will
(2) may NOT recommence unless 2FR1064 is OPERABLE

D. (1) will NOT
(2) may recommence if effluent flows are estimated once per 4 hours.

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that a low flow condition will automatically close the 2WL51. Incorrect in that loss of flow is not an enabled design feature with the radiation monitor.

B. Incorrect. Plausible because the operator may believe that the procedure requires both 2R18 and 2FR1064 operable in order to recommence the release and satisfy the requirements of the ODCM.

C. Incorrect. Plausible because the operator may believe that a low flow condition will automatically close the 2WL51. Plausible because the operator may believe that the procedure requires both 2R18 and 2FR1064 operable in order to recommence the release and satisfy the requirements of the ODCM.

D. Correct. Loss of discharge flow is not a design feature associated with the 2R18 process radiation monitor. Some process radiation monitors do have a feature that when programmed will isolate the process on low of flow. Per ODCM bases for liquid effluent (3.3.8), the controls and monitoring are determined to ensure that the 10 CFR 20 limits are not exceeded. Per Action 29 of the ODCM for an inoperable liquid radwaste effluent line monitor, the release may continue provided effluent flow is estimated once per 4 hours.

Technical References:	ODCM (R29), S2.OP-SO.WL-0001 (R32)
Proposed References to be provided:	None
Learning Objective:	NOS05CONDOP-14, Objective 5. Describe requirements for the following Control Room or Field Activities in accordance with applicable Conduct of Operations Manual Administrative Procedures
Question Source:	Bank – Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.11 / 43.4 / 45.10
Level:	RO
Category/System:	14-Radiation Control
K/A #:	G2.3.11 Ability to control radiation releases.
Importance:	3.8
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the candidate is being asked the bases of the ODCM administrative limits for the release of liquid radwaste and whether the release may proceed if actions are taken per the procedure/ODCM.

Additional References: Salem ODCM 3.3.8

PART I RADIOLOGICAL EFFLUENT CONTROLS — Salem REC ODCM Rev. 29 03/21

3/4.3 INSTRUMENTATION

3/4.3.3.8 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

TABLE 3.3-12: Radioactive Liquid Effluent Monitoring Instrumentation

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Gross Radioactivity Monitors Providing Automatic Termination of Release		
a. Liquid Radwaste Effluent Line	1 (Unit 1) - **1R18** 1 (Unit 2) - **2R18**	26
b. Steam Generator Blowdown Line	4 (Unit 1) - **1R19A-D** 4 (Unit 2) - **2R19A-D**	27
2. Gross Radioactivity Monitors not Providing Automatic Termination of Release		
a. Containment Fan Coolers – Service Water Line Discharge	2 (Unit 1) - **1R13A, B** 2 (Unit 2) - **2R13A, B**	28
b. Chemical Waste Basin	1 **(R37)**	31
3. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Line	1 (Unit 1) - **1FR1064** 1 (Unit 2) - **2FR1064**	29
b. Steam Generator Blowdown Line	4 (Unit 1) - **1FA-3178, -3180, -3182, -3184** 4 (Unit 2) - **2FA-3178, -3180, -3182, -3184**	29

Additional References: Salem ODCM 3.3.8 Bases

BASES

3/4.3 INSTRUMENTATION

3/4.3.3.8 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of the "old" 10 CFR Part 20 (ODCM Appendix F). The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

CROSS REFERENCE - TABLES 3.3-12 and 4.3-12

Unit 1 T/S Table Item No.	Instrument Description	Acceptable RMS Channels
1a	Liquid Radwaste Effluent Line Gross Activity	1R18
1b	Steam Generator Blowdown Line Gross Activity	1R19A, B, C, and D(1)
2a	Containment Fan Coolers Service Water Line Discharge Gross Activity	1R13A and B(1)

Unit 2 T/S Table Item No.	Instrument Description	Acceptable RMS Channels
1a	Liquid Radwaste Effluent Line Gross Activity	2R18
1b	Steam Generator Blowdown Line Gross Activity	2R19A,B,C, and D(1)
2a	Containment Fan Coolers - Service Water Line Discharge Gross Activity	2R13A and B(1)
2b	Chemical Waste Basin Line Gross Activity	R37

(1) The channels listed are required to be operable to meet a single operable channel for the ODCM's "Minimum Channels Operable" requirement.

69 **14-091** **Points: 1.00**

Given:

- You are the Primary Communicator

Then:

- A station event has just been classified as a Site Area Emergency

Which of the following describes the time requirement for State and Local notifications?

A. 30 minutes from the initiation of the event.

B. 15 minutes from the completion of the initial contact message form.

C. 15 minutes from the time the communicator acknowledges the briefing from the Shift Manager.

D. 15 minutes from the classification of the event.

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that the notification time includes the 15 minutes to classify and 15 minutes for the notification to the States. Incorrect in that the classification time can be less than 15 minutes, therefore 30 minutes for notifications is not always correct.

B. Incorrect. Plausible because the operator as the primary communicator usually waits until the ICMF is completed by the Shift Manager and is read to the communicator before notifying the States. Incorrect in that the notification time of 15 minutes starts when the Shift Manager declares the emergency classification.

C. Incorrect. Plausible because the operator as the primary communicator usually waits until the ICMF is completed by the Shift Manager and is read to the communicator before notifying the States. Incorrect in that the notification time of 15 minutes starts when the Shift Manager declares the emergency classification.

D. Correct. The 15 minute notification time to the States and Local agencies starts when the Shift Manager classifies or declares the emergency classification for the event.

Technical References:	EP-SA-325-F6 (R16)
Proposed References to be provided:	None
Learning Objective:	None
Question Source:	Bank – Beaver Valley 2017 RO75
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 43.3 / 45.11
Level:	RO
Category/System:	14-Emergency Procedures/Plan
K/A #:	G2.4.29 Knowledge of the emergency plan implementing procedures.
Importance:	3.1
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the applicant is being asked about Primary Communicator duties for time requirements to notify the States and Local agencies of the emergency classification.

70 **16-018** **Points: 1.00**

Given:

- Salem Unit 1 has been operating at 100% power for the past six months

Then:

- The reactor trips
- Shortly afterwards, a reactor startup is performed
- Twelve (12) hours after the trip, the startup has been paused with reactor power at 5%

Which of the following completes both statements?

1. Xenon-135 concentration will be __(1)__.
2. To maintain reactor power and T-avg stable over the next hour, the operator must add __(2)__ reactivity.

A. (1) rising (2) negative

B. (1) lowering (2) negative

C. (1) lowering (2) positive

D. (1) rising (2) positive

Answer: B

Explanation / Justification:

A. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible if the operator confuses the concept on how the fission product poison (xenon-135) effects reactivity and believes that a higher xenon concentration will require negative reactivity. Incorrect in that after 12 hours xenon has already peaked and xenon concentration will be decreasing adding positive reactivity, and therefore, require adding negative reactivity..

B. Correct. Directly following a reactor trip, the neutron flux is reduced essentially to zero. Therefore, Xe-135 is no longer produced directly from fission, but is no longer removed by burnup. The only remaining production mechanism is β-decay of the iodine-135; the only removal mechanism for xenon-135 is β-decay. Because the decay rate of iodine-135 is faster than the decay rate of xenon-135, the concentration builds to a peak (about -4700 pcm if tripped at 100% power). The time to peak is slightly less than the square root of the power from the trip (8-9 hours from 100%). Following the peak, the Xe-135 concentration will decrease at a rate controlled by the decay of Xe-135. Therefore, an additional twelve (12) hours after the power change, Xe-135 concentration will be decreasing due to the decay of I-135 and Xe-135. Additionally, returning to 5% power will result in a small amount of the burnout term removing Xe-135. A reduction in Xe-135 inserts positive reactivity (Xe-135 is a fission product poison). Therefore, to maintain s table reactor coolant temperature, the operator must insert negative reactivity because xenon-135 concentration is decreasing.

C. Incorrect. Part 1 is incorrect. Part 2 is correct. Plausible if the operator confuses the concept on how the fission product poison (xenon-135) effects reactivity and believes that a lower xenon concentration will require adding positive reactivity. Incorrect in that after 12 hours xenon has already peaked and xenon concentration will be decreasing adding positive reactivity, and therefore, require adding negative reactivity.

D. Incorrect. Part 1 is incorrect. Part 2 is incorrect. Plausible if the operator confuses the concept on how the fission product poison (xenon-135) effects reactivity and believes that a higher xenon concentration will require positive reactivity. Incorrect in that after 12 hours xenon has already peaked and xenon concentration will be decreasing adding positive reactivity, and therefore, require adding negative reactivity.

Technical References:	INPO GFES Fission Product Poisons (R3)
Proposed References to be provided:	None
Learning Objective:	NOS05POISON-02. INPO GFES Fission Product Poisons Rev. 3, ELO 1.4 Xenon Terms
Question Source:	Bank – NRC GFES Bank P3563
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.1
Level:	RO
Category/System:	16-Reactor Theory / Fission Product Poison
K/A #:	192006 K1.07 Describe the following processes and state their effect on reactor operations – xenon following a trip.
Importance:	3.4
Tier/Group	Theory-Tier 4

K/A Match: The K/A is a match because the candidate is being asked knowledge of how xenon-135 concentration is effecting reactivity following a reactor trip and what action the operator will need to take to compensate for the xenon.

Additional Reference: INPO GFES Fission Product Poison – Solution Bank

NATIONAL ACADEMY FOR NUCLEAR TRAINING **PWR GFE SOLUTION**	**REACTOR THEORY**	**P3563**
	192006 – FISSION PRODUCT POISONS	**K/A: K1.14 [3.2/3.3]**

A nuclear power plant had been operating at 100 percent power for two months when a reactor trip occurred. Soon afterward, a reactor startup was performed. Twelve hours after the trip, the startup has been paused with reactor power at 5 percent.

To maintain reactor power and reactor coolant temperature stable over the next hour, the operator must add _____ reactivity because the xenon-135 concentration will be _____.

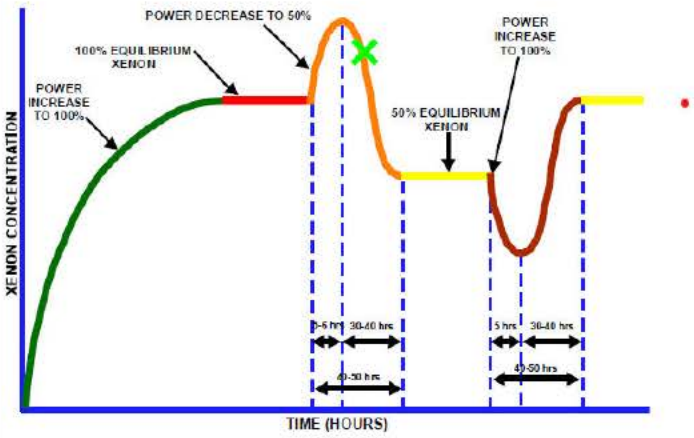
A. X positive; X increasing.

B. √ negative; X increasing.

C. X positive; √ decreasing.

D. √ negative; √ decreasing.

Discussion/Calculation:



Directly following a reactor trip, the neutron flux is reduced essentially to zero. Therefore, Xe-135 is no longer produced directly from fission, but is no longer removed by burnup. The only remaining production mechanism is the β- decay of the iodine-135; the only removal mechanism for xenon-135 is β- decay. Because the decay rate of iodine-135 is faster than the decay rate of xenon-135, the xenon concentration builds to a peak (about -4700 pcm if tripped from 100% power).

The time to "peak" is slightly less than the square root of the power from the trip (8-9 hours from 100% power). Following the peak, the Xe-135 concentration will decrease at a rate controlled by the decay of Xe-135.

Therefore, an additional twelve (12) hours after the power change, Xe-135 concentration will be *decreasing* due to the decay of I-135 and Xe-135. Additionally, returning to 5% power will result in a small amount of the burnout term removing Xe-135. A reduction in Xe-135 inventory inserts positive reactivity (Xe-135 is a fission product poison). *Therefore, to maintain a stable reactor coolant temperature, the operator must insert **negative** reactivity because xenon-135 concentration is **decreasing**.*

71 **16-019** **Points: 1.00**

Given:

- Salem Unit 2 was operating at 100% power when a Loss of All Offsite Power occurs.
- Due to plant complications, the Shift Manager has directed the crew to cooldown the RCS to Cold Shutdown per 2-EOP-TRIP-4, Natural Circulation Cooldown.
- The crew is borating the RCS for the RCS cooldown to Cold Shutdown.

Which of the following completes both statements?

1. As compared to adding boron to the RCS during forced circulation, adding boron during natural circulation requires __(1)__ time to achieve complete mixing in the RCS.
2. Once the RCS boron mixing is complete, a 1 ppm increase in RCS boron concentration during natural circulation will cause a/an __(2)__ change in reactivity that it would during forced circulation operation.

A. (1) less
(2) smaller

B. (1) less
(2) equal

C. (1) more
(2) smaller

D. (1) more
(2) equal

Answer: D

Explanation / Justification:

A. Incorrect. Part 1 is incorrect. Plausible if the operator misreads the question and therefore believes the question is asking about time to mix when forced circulation. Part 2 is incorrect. Plausible if the operator believes that reactivity effects will be smaller due to the lower RCS flowrates. Incorrect in that the change in reactivity will be the same when the boron is thoroughly mixed.

B. Incorrect. Part 1 is incorrect. Part 2 is correct. Plausible if the operator misreads the question and therefore believes the question is asking about time to mix when forced circulation

C. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible if the operator believes that reactivity effects will be smaller due to the lower RCS flowrates. Incorrect in that the change in reactivity will be the same when the boron is thoroughly mixed.

D. Correct. Due to loss of offsite power, no RCPs will be running resulting in a significantly lower RCS flowrate (NC conditions). Therefore, more time will be required to achieve complete mixing in the RCS for a given boron concentration change. Once the RCS mixing is complete, a 1 ppm increase (or decrease) in RCS boron concentration during NC operation will cause the same change in core reactivity that it would during forced circulation operation.

Technical References:	INPO GFES Fuel Depletion and Burnable Poisons (R3)
Proposed References to be provided:	None
Learning Objective:	NOS05POISON-02. Refer to INPO GFES Fuel Depletion and Burnable Poisons Rev. 3, ELO 1.6 Boron and Natural Circulation
Question Source:	Bank – NRC GFES Reactor Theory P3364
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.1
Level:	RO
Category/System:	16-Reactory Theory / Fuel Depletion and Burnable Poisons
K/A #:	192007 K1.05 Describe the effects of boration/dilution on reactivity during forced-flow and natural circulation conditions.
Importance:	3.2
Tier/Group	Theory-Tier 4

K/A Match: The K/A is a match because the candidate is being asked how boration is effected during natural circulation.

Additional Reference: INPO GFES Fuel Depletion and Burnable Poisons

ELO 1.6 Boron and Natural Circulation

Introduction

This section contains a short discussion of boron concentration changes during natural circulation conditions.

Boron Concentration Changes During Natural Circulation

Normally, boration and dilution are not performed during natural circulation conditions. Without the Reactor Coolant Pumps (RCPs) running the RCS flow rate will be much lower and the boron mixture distribution will be uneven throughout the core. This could lead to a loss of shutdown margin. To prevent a dilution accident, reducing boron concentration during natural circulation conditions is not normally allowed. Any boron changes in natural circulation will have the same reactivity effect as with forced circulation, once completely mixed.

The concentration of boron used for RCS boration remains the same throughout core life (from BOL to EOL) so the rate at which negative reactivity is added during boration does not change over core life. However, because the concentration of boron in the RCS decreases over core life by approximately a factor of 10, the rate at which dilution occurs (and thus, reactivity is added) is much slower (by approximately a factor of 10 as well).

So even though the reactor will respond quicker at the end of cycle (EOC) for a given reactivity positive addition from a boron dilution, the reactivity can be added quicker at the beginning of cycle (BOC) due to quicker changes in boron concentration.

72 **16-020** **Points: 1.00**

Given:

- Salem Unit 1 is performing a reactor startup with the reactor slightly subcritical.
- 2N31 and 2N32 Source Range count rates are stable.

Then:

- The RO performs a short control rod withdrawal.

Which of the following describes the response on the Source Range (SR) counts and startup rate after the rod withdrawal has stopped (assume the reactor remains subcritical)?

A. SR counts will slowly increase then gradually increase and stabilize at a higher value; startup rate will rapidly increase then gradually decrease to a slightly positive value.

B. SR counts will rapidly increase then stabilize at a higher value; startup rate will rapidly increase then gradually decrease and stabilize at zero.

C. SR counts will rapidly increase then gradually decrease and stabilize at the previous value; startup rate will rapidly increase then gradually decrease and stabilize at zero.

D. SR counts will rapidly increase then stabilize at a higher value; startup rate will not change until criticality is achieved.

Answer: B

Explanation / Justification:

A. Incorrect. First part is incorrect. Plausible because the operator may believe that the SR counts will continue to increase sometime after the rod pull has stopped. Incorrect in that this describes the SR count rate when the reactor is near critical conditions. Second part is incorrect. Plausible because the operator may believe that a stable positive SUR is necessary to achieve a higher equilibrium SR count rate. Incorrect in that in subcritical multiplication, SUR will initially rise then decay to zero with a new stable SR count rate.

B. Correct. The reactor will have a stable count rate in the SR while subcritical (keff < 1.0). When the operator stops rod motion in a subcritical reactor, the source range count rate will achieve a new higher equilibrium level due to the insertion of positive reactivity (keff increases, but still not critical). With each rod pull positive reactivity is inserted and the startup rate will increase then decay to zero indicating subcritical multiplication has reached equilibrium.

C. Incorrect. Plausible because the operator may believe that if the SUR is not constant, then the SR count rate will not achieve a higher value.

D. Incorrect. Plausible if the operator believes that the SR startup rate will not change until the reactor is critical.

Technical References:	INPO Reactor Operational Physics (R3)
Proposed References to be provided:	None
Learning Objective:	NOS05RXOPER-02. Refer to INPO Reactor Operational Physics Rev. 3, ELO 1.2 Reactor Startup Nuclear Instrumentation Response.
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.1
Level:	RO
Category/System:	16-Reactory Theory / Reactor Operational Physics (Startup and Approach to Criticality)
K/A #:	192008 K1.03 Describe count rate and instrument response that should be observed for rod withdrawal during the approach to criticality.
Importance:	4.0
Tier/Group	Theory-Tier 4

K/A Match: The K/A is matched because the operator is being asked how the SR count rate and startup rate behave during a subcritical reactor during a reactor startup.

Additional Reference: INPO GFES Solution Bank

NATIONAL ACADEMY FOR NUCLEAR TRAINING **PWR GFE SOLUTION**	**REACTOR THEORY**	**P265**
	192008 – REACTOR OPERATIONAL PHYSICS	**K/A: K1.03 [3.9/4.0]**

A reactor startup is in progress and the reactor is slightly subcritical in the source range. Assuming the reactor remains subcritical, a short control rod withdrawal will cause the reactor startup rate indication to increase sharply in the positive direction, and then...

A. X rapidly decrease and stabilize at a negative 1/3 dpm.

B. √ gradually decrease and stabilize at zero.

C. X stabilize until the point of adding heat (POAH) is reached; then decrease to zero.

D. X continue increasing until the POAH is reached; then decrease to zero.

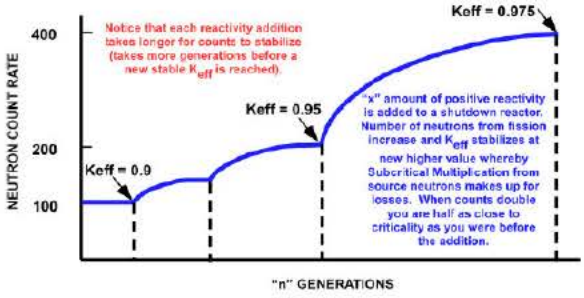
Discussion/Calculation:

A reactor will have a stable count rate in the source range while subcritical (k_{eff} < 1.0). Because k_{eff} < 1.0, each neutron generation will produce less fission neutrons than the previous generation; source neutrons make up for the losses each generation in fission neutrons to stabilize the count rate. *Subcritical multiplication* is the process of utilizing source neutrons and fuel (fission neutrons) to maintain constant neutron population with k_{eff} less than 1.0. If positive reactivity is inserted in the core (k_{eff} increases, but still not critical) the source range count rate will increase:

$$N_t = S_o\left(\frac{1}{1-K_{eff}}\right) \text{ or } N_t = S_o M$$

SUR indication during a rod withdrawal in a subcritical reactor looks like this:

1 – Jump Up (when rods movement started)
2 – Smile (during continuous rod withdrawal)
3 – Jump Down (when rod movement stopped)
4 – Decay to 0 as count rate stabilizes in the subcritical reactor.



Where, N_t = total neutrons, S_o = source neutrons, $1/(1-K_{eff})$ is referred to as the subcritical multiplication factor (M)

When positive reactivity is added to a subcritical reactor, an equilibrium condition occurs when fission neutron losses equal source neutron strength, causing startup rate (SUR) to initially increase then stabilize once the new steady state neutron count rate is reached (the stem indicates the reactor remains subcritical). The indication of SUR is basically the inverse of the graph above.

Additional Reference: INPO GFES Solution Bank

NATIONAL ACADEMY FOR NUCLEAR TRAINING **PWR GFE SOLUTION**	**REACTOR THEORY**	**P1065**
	192008 – REACTOR OPERATIONAL PHYSICS	**K/A: K1.03 [3.9/4.0]**

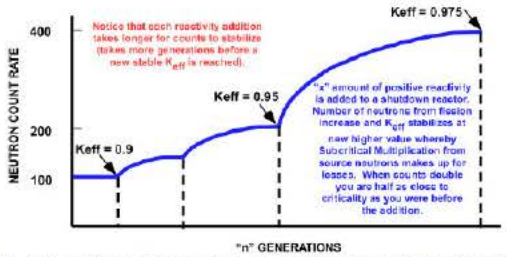
During a reactor startup, equal amounts of positive reactivity are being sequentially added, and the source range count rate is allowed to reach equilibrium after each addition. Which one of the following statements applies for each successive reactivity addition?

A. X The time required to reach equilibrium count rate is the same.
B. X The time required to reach equilibrium count rate is shorter.
C. √ The numerical change in equilibrium count rate is greater.
D. X The numerical change in equilibrium count rate is the same.

Discussion/Calculation:

A reactor will have a stable count rate in the source range while subcritical ($k_{eff} < 1.0$). Because $k_{eff} < 1.0$, each neutron generation will produce less fission neutrons than the previous generation; source neutrons make up for the losses each generation in fission neutrons to stabilize the count rate. *Subcritical multiplication* is the process of utilizing source neutrons and fuel (fission neutrons) to maintain constant neutron population with k_{eff} less than 1.0. If positive reactivity is inserted in the core (k_{eff} increases, but still not critical) the source range count rate will increase:

$$N_t = S_o\left(\frac{1}{1-K_{eff}}\right) \; or \; N_t = S_o M$$



Where, N_t = total neutrons, S_o = source neutrons, $1/(1-K_{eff})$ is referred to as the subcritical multiplication factor (M)

When positive reactivity is added to a subcritical reactor, an equilibrium condition occurs when fission neutron losses equal source neutron strength, causing startup rate (SUR) to initially increase then stabilize once the new steady state neutron count rate is reached (the stem indicates the reactor remains subcritical). Note that doubling the count rate by reactivity additions will reduce the margin to criticality by approximately one half.

It can be seen both in the equation and the graph above that as k_{eff} approaches 1.0 (criticality), the neutron population will increase at a much greater rate the closer the reactor is to criticality.

$$N_t = S_o\left(\frac{1}{1-K_{eff}\uparrow}\right) \rightarrow N_t \Uparrow$$

73 **16-021** **Points: 1.00**

Given:

- Salem Unit 2 is shutdown.
- RCS pressure is 1805 psig and stable.
- Core decay heat is being removed via the Steam Generators (SGs) using the MS10s, Atmospheric Dump Valves

What is the HIGHEST pressure in the SGs to obtain a MINIMUM 100 °F subcooling margin in the reactor coolant leaving the SGs (assume the reactor coolant leaves the SGs at the SG saturation temperature)?

A. 790 psig

B. 805 psig

C. 820 psig

D. 835 psig

Answer: B

Explanation / Justification:

A. Incorrect. Plausible if the operator incorrectly calculates the required subcooling temperature or incorrectly reads the Steam Table value.

B. Correct. At 1805 psig or 1820 psia the RCS saturation temperature is 623 °F. Therefore, to obtain a 100 °F subcooling margin in the RCS loop cold legs, temperature must be 100 °F lower than 623 °F (which is equal to 523 °F). Because a negligible temperature difference exist across the steam generator tubes, steam generator temperature must also be approximately 523 °F. Because core heat is being removed by the SGs, the SGs are a saturated system. Therefore, to determine correct SG pressure to obtain 100 °F subcooling margin, one must determine the saturation pressure at 523 °F, which is approximately 820 psia (which is equal to approximately 805 psig)..

C. Incorrect. Plausible if the operator fails to convert 820 psia to psig.

D. Incorrect. Plausible if the operator incorrectly calculates the required subcooling temperature or incorrectly reads the Steam Table value.

Technical References:	ASME Steam Tables
Proposed References to be provided:	ASME Steam Tables Rev. 2000
Learning Objective:	NOS05STEAM0-02.
Question Source:	Modified Bank – INPO GFES Bank P575
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.14
Level:	RO
Category/System:	16-Thermodynamics / Steam
K/A #:	193003 K1.24 Explain the usefulness of steam tables to the control room operator.
Importance:	3.1
Tier/Group	Theory-Tier 4

K/A Match: The K/A is matched because the operator is being asked to determine the saturation pressure in the steam generators that would allow for a 100 °.F subcooling in the RCS by using steam tables. This question shows the operators how useful steam tables can be when determining subcooling.

Parent Question:

NATIONAL ACADEMY FOR NUCLEAR TRAINING **PWR GFE SOLUTION**	**THERMODYNAMICS**	**P575**
	193003 – STEAM	**K/A: K1.17 [3.0/3.2]**

A reactor is shut down with reactor coolant system (RCS) pressure at 1,500 psia and core decay heat is being removed via the steam generators (SGs). What pressure must be maintained in the SGs to obtain a 110°F subcooling margin in the reactor coolant leaving the SGs? (Assume the reactor coolant leaves the SGs at the SG saturation temperature.)

A. X 580 psia

B. √ 600 psia

C. X 620 psia

D. X 640 psia

74 **16-022** **Points: 1.00**

Given:

- Salem Unit 2 is operating at 80% power.
- All Circulators are in service.

Which of the following completes the statement?

If condensate depression decreases from 5 °F to 2 °F, the steam cycle thermal efficiency will __(1)__ and the condensate pumps will operate __(2)__ cavitation.

A. (1) lower (2) closer to

B. (1) lower (2) farther from

C. (1) rise (2) closer to

D. (1) rise (2) farther from

Answer: C

Explanation / Justification:

A. Incorrect. First part is incorrect. Plausible because the operator may believe with a lower condensate depression, the feedwater temperature will lower causing a rise in reactor power, and therefore, increase in plant efficiency. Second part is correct.

B. Incorrect. First part is incorrect. Plausible because the operator may believe with a lower condensate depression, the feedwater temperature will lower causing a rise in reactor power, and therefore, increase in plant efficiency. Second part is incorrect. Plausible because the operator may incorrectly recall how a lower condensate depression will affect suction pressure at the condensate pump and the likelihood of pump cavitation. Incorrect in that a lower condensate depression will increase the likelihood of pump cavitation.

C. Correct. Changing the condensate depression from 5 °F to 2 °F will decrease the condensate depression. If condensate depression decreases, then the condensate will be closer to saturation temperature. This produces less NPSH at the suction of the condensate pump, raising the probability of cavitation. However, because the condensate is now at a higher temperature, the steam generators must add less sensible heat to the feedwater to reach saturated conditions, increasing plant efficiency.

D. Incorrect. First Part is correct. Second Part is incorrect. Plausible because the operator may incorrectly recall how a lower condensate depression will affect suction pressure at the condensate pump and the likelihood of pump cavitation. Incorrect in that a lower condensate depression will increase the likelihood of pump cavitation.

Technical References:	INPO GFES Thermodynamic Process (R3)
Proposed References to be provided:	None
Learning Objective:	NOS05THRMPR-03. Refer to INPO Thermodynamic Process Rev. 3, ELO 2.2 Condenser Design and Characteristics
Question Source:	Bank – INPO GFES P2576
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.14
Level:	RO
Category/System:	16-Thermodynamics / Thermodynamic Process
K/A #:	1930004 K1.11 Describe the process of condensate depression (subcooling) and its effect on plant operation.
Importance:	2.5
Tier/Group	Theory-Tier 4

K/A Match: The K/A is matched because the operator is being asked about the concept of condensate depression and how it impacts plant operation.

Additional Reference:

NATIONAL ACADEMY FOR NUCLEAR TRAINING PWR GFE SOLUTION	THERMODYNAMICS	P2576
	193004 – PROCESSES	K/A: K1.11 [2.4/2.5]

A nuclear power plant is operating at 80 percent power with 5ºF of condensate depression in the main condenser. If the condensate depression decreases to 2ºF, the steam cycle thermal efficiency will _____; and the condensate pumps will operate _____cavitation.

A. √ increase; √ closer to

B. √ increase; X farther from

C. X decrease; √ closer to

D. X decrease; X farther from

Discussion/Calculation:

Condensate depression is defined as the temperature difference between the saturation temperature (for the condenser pressure) and the temperature of the condensate. It is the process whereby the water droplets cling to the tube and lose more heat than the latent heat of condensation. They become subcooled and denser then drop off the tubes into the hotwell.

Condensate depression decreases the overall plant efficiency because more sensible heat must be added to reach saturated conditions in the steam generators. However, the advantage of having a small degree of condensate depression is to ensure adequate net positive suction head (NPSH) for the main condensate pumps and prevent cavitation. Excessive condensate depression causes more air absorbed in condensate and accelerates oxygen corrosion of secondary plant materials.

If condensate depression decreases, then the condensate will be closer to saturation temperature. This produces less NPSH at the suction of the condensate pump, raising the probability of cavitation. However, because the condensate is now at a higher temperature, the steam generators must add less sensible heat to the feedwater to reach saturated conditions, increasing plant efficiency.

At a higher temperature, plant efficiency will increase because less sensible heat must be added in the steam generators. However, probability of cavitation will increase as subcooling (or condensate depression decreases).

Answer/Distractor Analysis:

A. CORRECT. (2/2) See discussion above.

B. WRONG. (1/2) The probability of cavitation will increase as subcooling (or condensate depression decreases).

C. WRONG. (1/2) At a higher temperature, plant efficiency will increase because less sensible heat must be added in the steam generators

D. WRONG. (0/2) At a higher temperature, plant efficiency will increase because less sensible heat must be added in the steam generators. However, probability of cavitation will increase as subcooling (or condensate depression decreases).

75 **16-023** **Points: 1.00**

Given:

- Salem Unit 1 is operating at 80% power, middle of life of the fuel cycle.
- All control rods are withdrawn except D bank is at 190 steps and in manual control.
- Core axial power distribution is peaked below the core midplane.

Which of the following will **INCREASE** the core maximum axial peaking factor? (Assume no operator action is taken unless stated and that main turbine load and core xenon distribution do not change unless stated

A. Turbine load/reactor power is reduced by 10 percent.

B. The control bank D is withdrawn 4 steps.

C. Reactor coolant system boron concentration is reduced by 15 ppm.

D. A fully withdrawn control rod located at the edge of the core drops to the bottom of the core.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that the load reduction will lower hot leg temperatures causing the flux in this portion to increase and resulting in an increase in the axial peaking factor. Incorrect in that the lower hot leg temperature will result in a higher density causing the neutron flux to shift higher in the core due to the increased neutron moderation in these regions (DT gets smaller as power id reduced). This will reduce the axial peaking factor as the flux in the lower potions of the core lowers relative to the average core flux.

B. Incorrect. Plausible because the operator may believe that withdrawing control rods 4 steps will increase the hot leg temperature resulting in a higher flux in this portion and higher peaking factors. Incorrect in that withdrawing one bank of control rods 4 steps will cause the flux to be depressed less (with no change in average flux). This results in a decrease in axial peaking factor.

C. Correct. Reducing boron concentration (diluting) by 15 ppm will insert positive reactivity in the core, resulting on both heat balance (calorimetric) reactor power and average coolant temperature increasing. Also, diluting raises hot leg temperature, making the water in the upper regions of the core less dense (less neutron moderation), thereby shifting the flux to the already flux-dense portion below midplane. Another way to think about this is that the water is colder in the bottom of the core resulting in a greater change in the atoms per cubic centimeter being removed in the bottom of the core. This again adds positive reactivity in the bottom of the core and will shift the flux downward.

D. Incorrect. Plausible because the operator may believe that one control rod dropped in the core will increase the flux in that portion of the core adjacent to the dropped rod resulting in a higher axial peaking factor. Incorrect in that one control rod inserting will affect radial flux peaking, but will have no effect on axial flux peaking because the overall axial flux profile does not shift significantly since it is an axially uniform poison.

Technical References:	INPO GFES Core Thermal Limits (R3)
Proposed References to be provided:	None
Learning Objective:	NOS05CORTHR-02. Refer to INPO GFES Core Thermal Limits Rev. 3. ELO 1.3 Reactor Operation Effects on Peaking Factors.
Question Source:	Bank – INPO GFES P7650
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.14
Level:	RO
Category/System:	16-Thermodynamics / Core Thermal Limits
K/A #:	193009 K1.07 Describe factors that affect peaking and hot channel factors.
Importance:	3.3
Tier/Group	Theory-Tier 4

K/A Match: The K/A is matched because the operator is being asked which plant conditions will affect the axial peaking (or hot channel) factor.

Additional Reference: INPO GFES Core Limits Solution Bank

NATIONAL ACADEMY FOR NUCLEAR TRAINING PWR GFE SOLUTION	THERMODYNAMICS	P7650
	193009 – CORE THERMAL LIMITS	K/A: K1.02 [2.3/2.8]

A reactor is operating at 80 percent power near the middle of a fuel cycle. All control rods are nearly fully withdrawn and in manual control. Core axial power distribution is peaked below the core midplane.

Which one of the following will increase the core maximum axial peaking (or hot channel) factor? (Assume no operator action is taken unless stated, and that main turbine load and core xenon distribution do not change unless stated.)

A. X Turbine load/reactor power is reduced by 10 percent.

B. X The controlling bank of control rods is withdrawn 4 inches.

C. √ Reactor coolant system boron concentration is reduced by 15 ppm.

D. X A fully withdrawn control rod located at the edge of the core drops to the bottom of the core.

Discussion/Calculation:

The steady-state axial peaking (or hot channel) factor is a measure of maximum total power produced in fuel rod, ensuring enthalpy rise in the *entire* core is low enough to preclude reaching DNB. Measurements for this parameter can be obtained using in-core detectors taking a series of measurements across the full length of the core, and determining the *ratio of average flux for one elevation to the average flux for all elevations*. This factor depends on several parameters including fuel loading, control rod bank insertion, and fuel burn-up.

Reducing boron concentration (diluting) by 15 ppm will insert positive reactivity in the core, resulting in both heat balance (calorimetric) reactor power and average coolant temperature increasing. Also, diluting raises hot leg temperature, making the water in the upper regions of the core less dense (less neutron moderation), thereby shifting the flux to the already flux-dense portion below midplane. Another way to think about this is the fact that the water is colder in the bottom of the core resulting in a greater change in the atoms per cubic centimeter being removed in the bottom of the core. This again adds positive reactivity in the bottom of the core and will shift the flux downward. Therefore, Choice "C" is correct.

Answer/Distractor Analysis:

A. WRONG. Reducing reactor power by 10% results in a significant reduction in hot leg temperature with a relatively constant cold leg temperature. As hot leg temperature is reduced, water in the upper portions of the core is at a reduced temperature, resulting in a higher density, causing the neutron flux to shift higher in the core due to increased neutron moderation in these regions (Delta-T gets smaller as power is reduced). This will reduce the axial peaking factor as the flux in the lower portions of the core lowers relative to the average core flux.

B. WRONG. The stem indicates that the core axial distribution is already peaked below the core midplane. Withdrawing one bank of control rods 4 inches will cause the flux to be depressed less (with no change in average flux). This results in an decrease in axial peaking factor.

C. CORRECT. See discussion above.

D. WRONG. One control rod fully inserting will affect radial flux peaking, but will have no affect on axial flux peaking because the overall axial flux profile does not shift significantly since it is an axially uniform poison.

SRO EXAMINATION ANSWER KEY

SALEM 2023 NRC Exam (21-01 ILOT) **– (Rev. 1)**

76 **10-144** **Points: 1.00**

Given:

- Salem Unit 2 experienced an automatic reactor trip and Safety Injection due to lowering RCS pressure
- The crew suspects a PZR vapor space leak

Then:

- The crew is implementing 2-EOP-LOCA-1, Loss of Reactor or Secondary Coolant
- Subcooling is 10 °F
- RCS pressure is 1450 psig and stable
- PZR level is 30% and rising

Based on the above conditions, which of the following completes both statements?

1. Which ECCS Pump(s) is/are injecting?

2. Per 2-EOP-LOCA-1, what procedure will the CRS direct NEXT?

A. (1) Charging and SI
(2) 2-EOP-LOCA-2, Post LOCA Cooldown and Depressurization

B. (1) Charging ONLY
(2) 2-EOP-TRIP-3, SI Termination

C. (1) Charging and SI
(2) 2-EOP-TRIP-3, SI Termination

D. (1) Charging ONLY
(2) 2-EOP-LOCA-2, Post LOCA Cooldown and Depressurization

Answer: C

Explanation / Justification:

A. Incorrect. First part is correct. Second part is incorrect. Plausible because you would ultimately end up on 2-EOP-LOCA-2. Incorrect in that it would not be the next procedure path to transition to.

B. Incorrect. First part is incorrect. Plausible if the operator incorrectly recalls the shutoff head for an SI pump. Second part is correct.

C. Correct. Basically the conditions in the stem describes a small break LOCA and the conditions will meet SI Termination criteria in EOP-LOCA-1 and the SRO will transition to EOP-TRIP-3 (SI Termination). In EOP-TRIP-3, the crew will reduce ECCS injection flow, which with the unisolable PZR vapor space leak (SBLOCA), RCS pressure will lower when ECCS injection flow is reduced. EOP-TRIP-3 will then transition the crew to EOP-LOCA-2. RCS pressure at 1450 psig, will be above the shutoff head for the SI and Charging Pumps and will both be injection flow into the RCS.

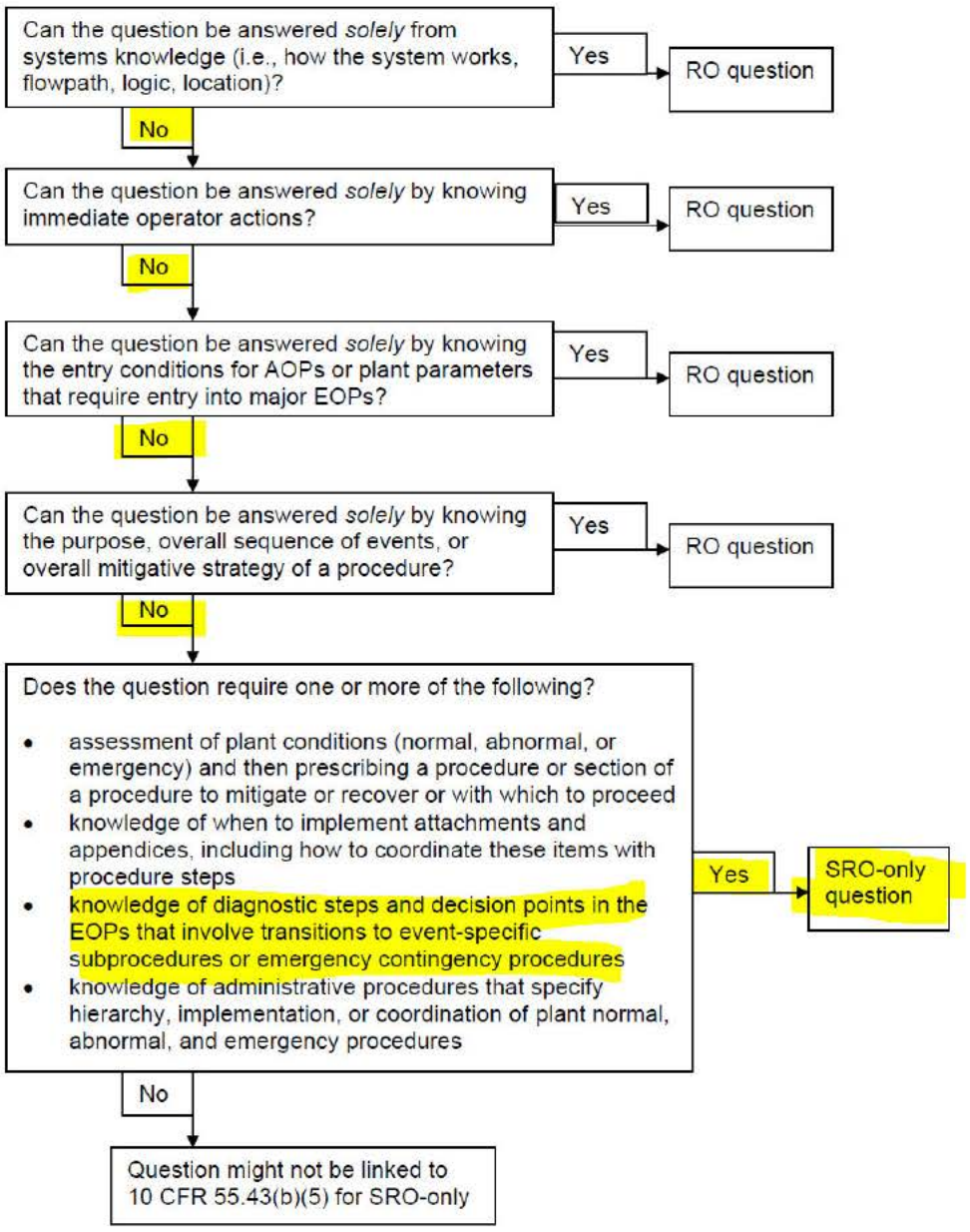
D. Incorrect. First part is incorrect. Second part is incorrect.

Technical References:	2-EOP-LOCA-1 (R42), 2-EOP-LOCA-2 (R41), 2-EOP-TRIP-3 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05LOCA01-07, Objective 3. For the following analyzed transients/accidents: Inadvertent Depressurization of the RCS - Stuck Open Safety (Condition II - Faults of Moderate Frequency. Describe/State the following: Describe the expected plant response
Question Source:	Modified Bank – Callaway 2021 NRC SRO76
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	SRO
Category/System:	11-Pressurizer Vapor Space Accident
K/A #:	APE 8 AA2.15 Ability to determine and/or interpret the following as they apply to the Pressurizer Vapor Space Accident: ECCS status.
Importance:	3.9
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the candidate is being asked to determine how ECCS flow injection will behave following SI actuation and RCS pressure due to a vapor space leak and what procedure will the SRO direct based on RCS pressure checks in EOP-LOCA-1.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Parent Question: Callaway 2021 NRC SRO76

Modified Question: Part 1 of the question was changed to address the K/A part on ECCS status. The question is asking the applicant to determine the status of ECCS flow(s) following the SI actuation during a PZR PORV Vapor Space accident and then what procedure will the CRS direct next based on the conditions in the stem.

Question # 76

An automatic reactor trip and safety injection has occurred as a result of lowering RCS pressure.

The crew has just entered E-1, Loss of Reactor or Secondary Coolant and plant conditions are as follows:

	Prior to SI	After SI Occurs
Pzr Pressure	Lowering	500 psig and Lowering
RCS Temperature	Stable	Stable
Pzr Level	Stable	Rising

(1) What accident would result in these conditions?

And

(2) The CRS will direct what procedure NEXT?

A. (1) RCS Cold Leg Leak
(2) ES-1.1, SI Termination

B. (1) RCS Cold Leg Leak
(2) ES-1.2, Post LOCA Cooldown and Depressurization

C. (1) PZR Vapor Space Leak
(2) ES-1.1, SI Termination

D. (1) PZR Vapor Space Leak
(2) ES-1.2, Post LOCA Cooldown and Depressurization

Answer: D

77 **10-145** **Points: 1.00**

Given:

- A LOCA is in progress on Salem Unit 2.
- The crew is implementing 2-EOP-LOCA-1, Loss of Reactor or Secondary Coolant, and is currently at Step 7.4 for assessing conditions for stopping containment spray pumps.
- Containment pressure is 23 psig.

Then:

- A valid PURPLE path exists on Thermal Shock
- Containment pressure is below 13 psig.

In accordance with OP-SA-108-101-2002, EOP Users Guide, which of the following is required and why?

Note: FRP = Functional Restoration Procedure(s)

A. Step 7.4 may be performed if the CRS determines there is no conflict with implementing the FRP.

B. Immediately perform Step 7.4; a Note step in EOP-LOCA-1 specifically states NO FRPs are to be implemented during performance of EOP-LOCA-1.

C. Immediately perform Step 7.4; Step 7.4 is a continuous action step and still in effect.

D. DO NOT perform Step 7.4; transition to FRP for PURPLE path.

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that if a conflict arises between actions of an EOP and FRP that the CRS will always have the option to determine the best path to take, for example "Thinking Compliance".

B. Incorrect. Plausible because EOP-LOCA-3 has a Note (N1-1) that does state to NOT implement FRPs while performing actions to establish cold leg recirculation.

C. Incorrect. Plausible because step 7.4 is a continuous action step and would be performed when conditions are met. Incorrect in that continuous actions are not performed when transitions are made to a Red or Purple path FRP.

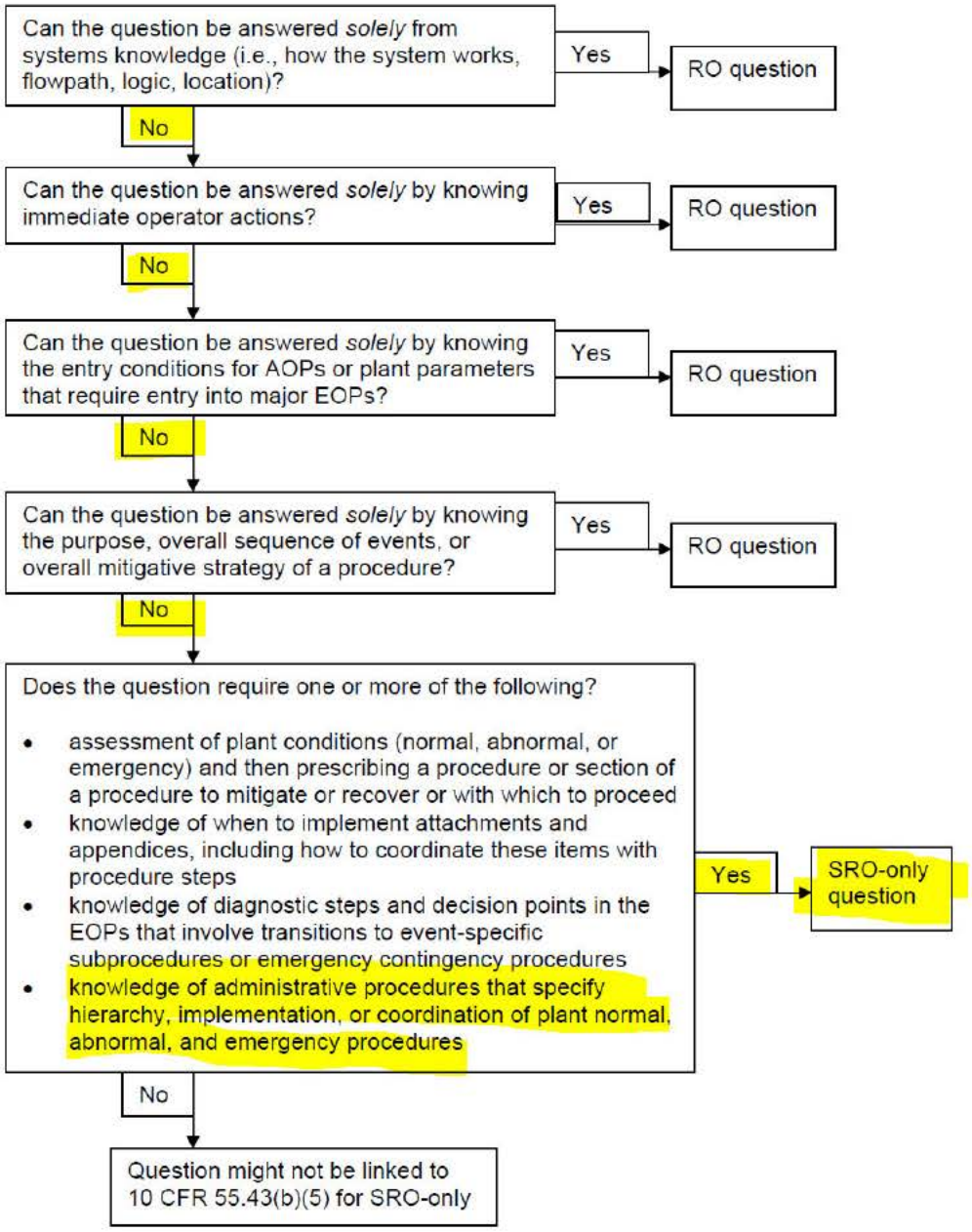
D. Correct. Per OP-SA-108-101-2002 step 2.16.2, a continuous action step should not be performed when a Red or Purple path FRP is in progress. Therefore, even though conditions of LOCA-1 step 7.4 are met, the crew should not stop CS pumps.

Technical References:	OP-SA-108-101-2002 (R0), 2-EOP-CFST-1 (R41)
Proposed References to be provided:	**Embedded step from EOP-LOCA-1**
Learning Objective:	NOS05TRP001-09, Objective 10. Given EOP-CFST-1 and a set of plant conditions, determine whether Functional Restoration Procedures (FRPs) should be implemented and/or terminated, in accordance with OP-SA-108-101-2002 and EOP-CFST-1
Question Source:	Bank - Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 43.1 / 45.13
Level:	SRO
Category/System:	10-Large Break LOCA / Emergency Procedures/Plan
K/A #:	EPE 11 G2.4.14 Knowledge of general guidelines for emergency and abnormal operating procedures usage
Importance:	4.5
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the candidate is being asked the rules of EOP usage when a Red or Purple path FRP exists and which procedure takes priority when a CAS action condition in the EOP is met.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Additional Reference: OP-SA-108-101-2002

2.16 Continuous Action Steps:

2.16.1. Most guidelines have been provided with a Continuous Action Summary. This is annotated in the upper left corner of each EOP Flowchart page. The particular items of the Continuous Action Summary associated with any guideline are provided in the FOLDOUT PAGE ITEMS document of the Generic Issues Section of the Westinghouse ERG Executive Volume. These foldout page items should be applied to the plant specific EOPs on a per guideline basis. The Continuous Action Summary contains several pieces of information or actions which are normally applicable at any step of the particular guideline. The most important of these actions are guideline transitions which allow immediate response to new symptoms as they appear. The placement of these transitions in the Continuous Action Summary allows prompt response to the appearance of subsequent symptoms.

2.16.2. Certain operator actions provided in a guideline direct or imply continuous performance throughout the remainder of the guideline. A continuous action is an action that is applicable from the point at which it is first encountered until superseded by alternate guidance or stated to be inapplicable. A continuous action generally remains applicable throughout its associated guideline unless otherwise stated, and may apply after a transition is made to another guideline if it does not contain any actions that are inappropriate for the subsequent guideline. For example, a transition contained within a continuous action will not apply after a transition is made to a RED or PURPLE path Functional Restoration Guideline or the LOPA series. Furthermore, actions requiring equipment to be started should not be performed in the LOPA series unless directed in that series. Additionally, Optimal Recovery Guideline actions (EOPs) are not to be performed while a Critical Safety Function is being restored from a RED or PURPLE condition.

78 **11-196** **Points: 1.00**

Given:

- Salem Unit 2 is operating at 30% power
- OHA D-36, RCP VIB HI, is LIT
- 21 RCP parameters are:
  - Motor flange vibration is 3 mils and steady
  - Shaft vibration is 16 mils and steady
- Engineering has determined the vibrations are valid
- The crew enters S2.OP-AB.RCP-0001, RCP Abnormality

What action will the CRS direct?

Note: TS 3.4.1.1, Reactor Coolant Loops and Coolant Circulation
2-EOP-TRIP-1, Reactor Trip or Safety Injection

A. Trip the reactor and stop 21 RCP per Attachment 2, Stopping RCPs, of S2.OP-AB.RCP-0001.

B. Preparations for a controlled Unit shutdown to remove 21 RCP from service.

C. Stop 21 RCP within 1 hour per TS 3.4.1.1 and place the Unit in MODE 3 within 6 hours.

D. Trip the reactor and stop 21 RCP per OHA D-36 Alarm Response Procedure and enter 2-EOP-TRIP-1.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because motor and flange vibrations are parameters on Attachment 1 of S2.OP-AB.RCP-0001 that may require stopping RCPs per Attachment 2. Incorrect in that the Attachment 1 RCP vibration levels have not been exceeded as of yet.

B. Correct. The RCP vibration levels are below the requirements in Attachment 1 of AB.RCP-1 that require stopping the affected RCP per Attachment 2. The OHA ARP for D-36 directs entry into S2.OP.AB-RCP-0001. Per S2.OP-AB.RCP-0001 Step 3.10 allows the CRS/SM to determine if a Unit shutdown is required.

C. Incorrect. Plausible if the applicant believes that since RCP trip criteria has not been reached per Attachment 1 of S2.OP-AB.RCP-0001, then take the action of TS 3.4.1.1 for a loss of an RCP. Incorrect in that the TS action is to place the unit in MODE 3 within one hour on a loss of RCS flow.

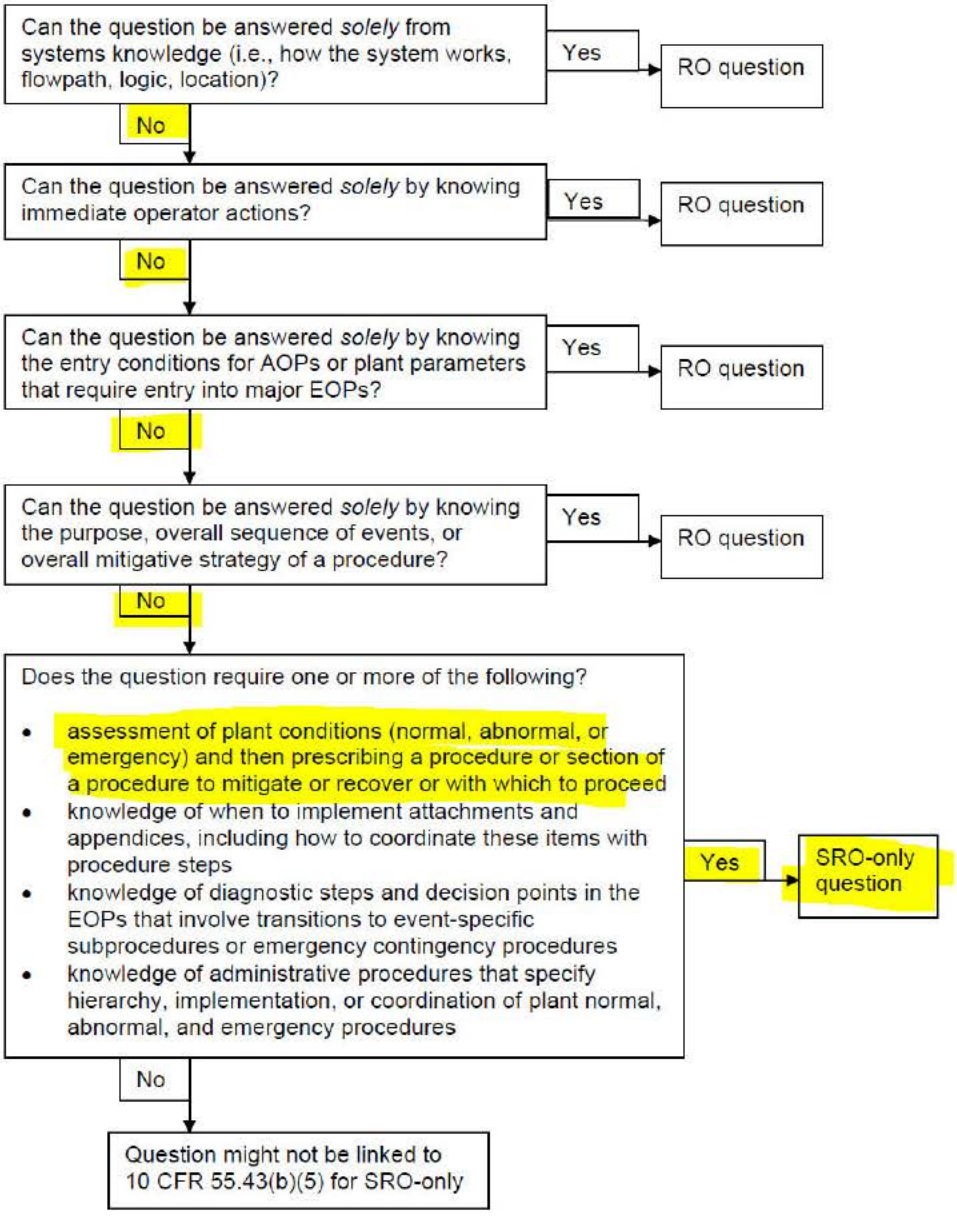
D. Incorrect. Plausible if the applicant believes that the ARP will direct stopping and tripping the reactor which it does not, only directs entry into S2.OP-AB.RCP-0001

Technical References:	S2.OP-AB.RCP-0001 (R28), TS 3.4.1.1.
Proposed References to be provided:	None
Learning Objective:	NOS05ABRCP0-08, Objective 4b. Given a set of initial plant conditions: Describe the final plant condition that is established by the abnormal procedure.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	SRO
Category/System:	11-Reactor Coolant Pump Malfunctions
K/A #:	APE 15 AA2.16 Ability to determine and/or interpret the following as they apply to RCP Malfunctions: RCP vibration
Importance:	3.5
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the candidate is being asked to interpret RCP vibrations and determine the required course of action per S2.OP-AB.RCP-0001.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Additional Reference: S2.OP-AB.RCP-0001

ATTACHMENT 1
(Page 1 of 2)

CONTINUOUS ACTION SUMMARY

___ 1.0 IF AT ANY TIME, any of the following validated RCP conditions exist, THEN GO TO Attachment 2, Stopping Reactor Coolant Pumps.

_____ Time

- Component Cooling Water flow is lost to ALL RCPs.
- Component Cooling Water flow not restored within 5 minutes of initial loss:
  - OHA-D-20, 21 RCP BRG CLG WTR FLO LO
  - OHA-D-21, 22 RCP BRG CLG WTR FLO LO
  - OHA-D-22, 23 RCP BRG CLG WTR FLO LO
  - OHA-D-23, 24 RCP BRG CLG WTR FLO LO
- RCP Seal Injection Flow AND RCP Thermal Barrier Component Cooling flows are lost concurrently (RCPs should be secured within 2 minutes to prevent RCP damage).
- Shaft vibration greater than 20 mils or motor flange vibration greater than 5 mils [C0379]
- Motor Bearing temperature greater than 175°F
- Motor Winding temperature greater than 302°F, (Modes 1-5)
  (Also, **REFER TO** Attachment 3 (page 2 of 2) for Alternate Indications to validate indicated rise in RCP Motor Winding Temperature)
- Seal Water Outlet temperature greater than 190°F
- #1 Seal Water Leakoff flow less than 0.8 gpm and reverses to greater than or equal to 6.0 gpm or pump bearing/seal inlet temperature rise
- #1 Seal Water Leakoff flow less than 0.8 gpm with rising pump bearing/seal inlet temperatures
- #1 Seal Water Leakoff flow greater than or equal to 6 gpm

Additional Reference:

___ 3.9 IF the RCP malfunction is NOT determined,
THEN **RETURN** to Step 3.1.

NOTE
___ Following a Loss of Seal Injection event, with RCPs having a seal leakoff less than 2.5 gpm, RCS cooldown will help to lower seal leakoff temperatures.

___ 3.10 IF Unit shutdown is determined by CRS/SM,
THEN:

___ A. **INITIATE** Unit shutdown IAW applicable Integrated Operating Procedures.

___ B. **NOTIFY** CRS/SM to refer to the following:

- Technical Specifications
- Event Classification Guide

G. Step 3.10, provides direction to perform a controlled Unit shutdown when unacceptable RCP conditions exist. The unit is shutdown in a period of 8 hours or less and the affected RCP(s) are removed from service allowing a more detailed inspection and analysis. Notification for CRS/SM to evaluate plant conditions. Plant Technical Specifications and the Event Classification Guide are evaluated for applicability.

79 **14-081** **Points: 1.00**

Given:

- Salem Unit 2 is in MODE 5
- 21 RHR Pump is in shutdown cooling
- 22 RHR Pump is in standby

Then:

- 21 RHR Pump trips
- The crew enters S2.OP-AB.RHR-0001, Loss of RHR
- Attempt to start 22 RHR Pump results in the breaker tripping free
- STA reports Time to Core Boiling is less than 15 minutes
- Field operator reports no protective relay flags are present and no obvious reason for the 4KV breaker for 22 RHR Pump to trip free, and an acrid smell is observed near the 21 RHR 4KV breaker cubicle

In accordance with OP-SA-108-106-1001, Equipment Control and Large Motor Starting Criteria, complete the following statement concerning breaker re-closure for 22 RHR Pump?

Based on the above conditions, the __(1)__ can authorize one attempt at reclosing the breaker without an investigation ONLY during __(2)__.

A. (1) Shift Manager
(2) emergency conditions

B. (1) Plant Manager
(2) emergency conditions

C. (1) Shift Manager
(2) emergency conditions AND with Shift Technical Advisor concurrence

D. (1) Plant Manager
(2) emergency conditions AND with Shift Technical Advisor concurrence

Answer: A

Explanation / Justification:

A. Correct. Per OP-SA-108-106-1001 step 4.2.2, if an emergency condition exists, as determined by the SM/CRS, and a breaker fails to close, one attempt at reclosure may be made without conducting an investigation, if that piece of equipment is essential for maintaining plant stability.

B. Incorrect. Plausible because the conditions in the stem are severe and the SRO may believe that the Plant Manager level of management is required to accept responsibility if the RHR motor is damaged.

C. Incorrect. Plausible because the SRO may think that before reclosing the breaker to an important safety equipment that the on-shift Shift Technical Advisor should be involved in the decision process.

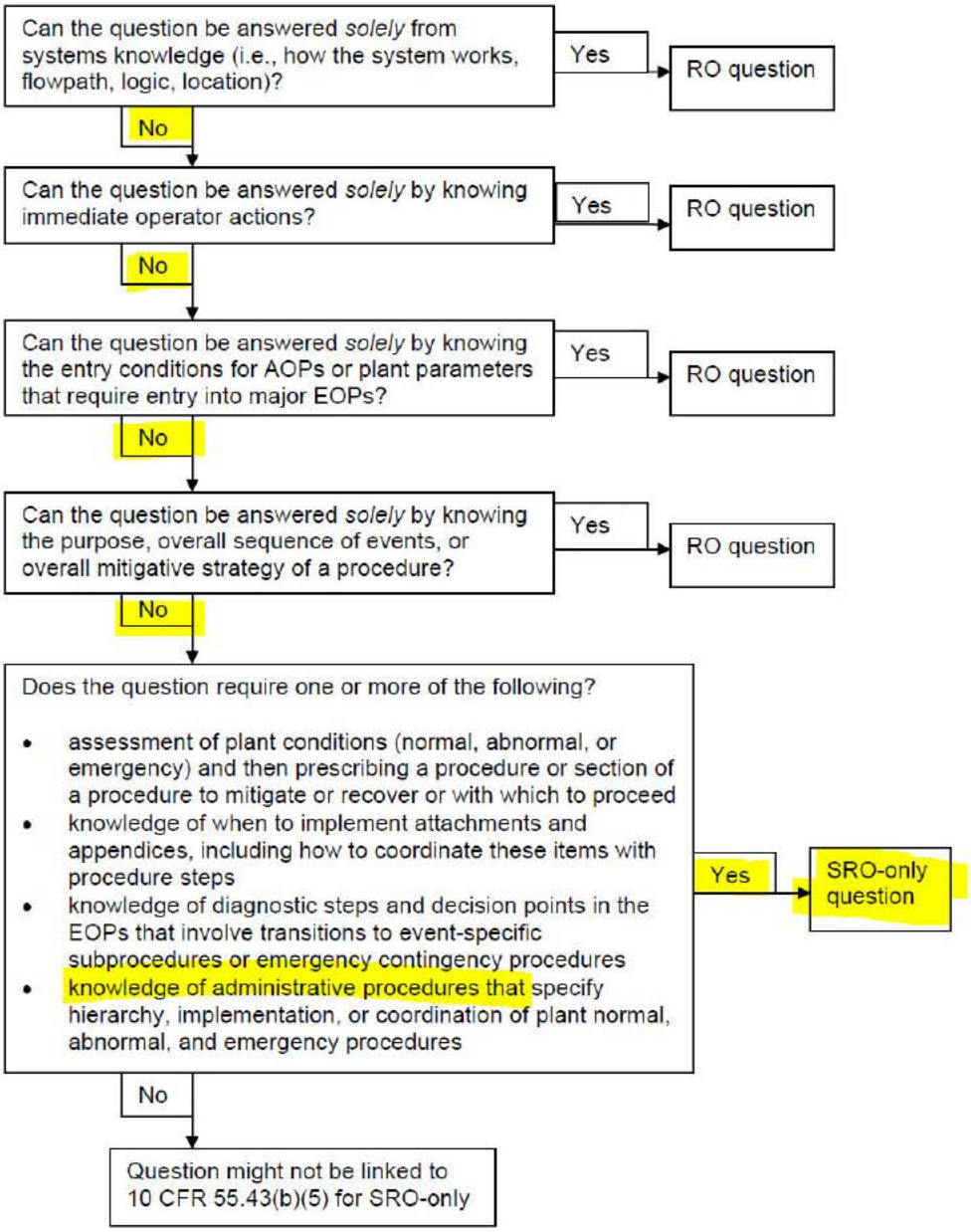
D. Incorrect. Plausible because the SRO may think that before reclosing the breaker to an important safety equipment that the on-shift Shift Technical Advisor should be involved in the decision process.

Technical References:	OP-SA-108-106-1001 (R1)
Proposed References to be provided:	None
Learning Objective:	NOS05CONDOP-15, Objective 5. Describe requirements for the following Control Room or Field Activities in accordance with applicable Conduct of Operations Manual Administrative Procedures
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.10 / 45.13
Level:	SRO
Category/System:	14-Loss of RHR System / Conduct of Operations
K/A #:	APE 25 G2.1.1 Knowledge of conduct of operations requirements
Importance:	4.2
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the candidate is being ask knowledge of the breaker reclosure policy when the breaker trips free while attempting to start an RHR pump during loss of RHR cooling.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



80 **10-146** **Points: 1.00**

Given:

- Salem Unit 2 reactor failed to automatically trip following a valid RPS trip signal
- All attempts to trip the reactor from the main control room have been unsuccessful
- The crew enters 2-EOP-FRSM-1, Response to Nuclear Power Generation, and all immediate actions are complete

In accordance with 2-EOP-FRSM-1, which of the following completes both statements?

1. Which of the following describes how the turbine trip status is checked?
2. What subsequent action will the CRS direct if the turbine trip status check is NOT met following immediate actions of 2-EOP-FRSM-1?

A. (1) ALL turbine stop valve closed bi-stables LIT on 2RP4.
(2) Dispatch an operator to locally trip the turbine at the front standard.

B. (1) ALL turbine stop valve closed bi-stables LIT on 2RP4.
(2) Direct the control room operator to stop all EHC pumps.

C. (1) 2 out 3 Auto Stop Oil low pressure bi-stables LIT on 2RP4.
(2) Direct the control room operator to stop all EHC pumps.

D. (1) 2 out 3 Auto Stop Oil low pressure bi-stables LIT on 2RP4.
(2) Dispatch an operator to locally trip the turbine at the front standard.

Answer: A

Explanation / Justification:

A. Correct. Per 2-EOP-FRSM-1 Step 6, the turbine is tripped when all stop valves are closed. This can be verified by observing the bi-stables on 2RP4. If the turbine fails to trip at Step 6, then the CRS will dispatch an operator to the turbine deck to locally trip the turbine at the front standard.

B. Incorrect. First part is correct. Second part is incorrect. Plausible because stopping EHC pumps is possible from the control room. Incorrect in that FRSM-1 step 6 directs locally tripping the turbine at the front standard.

C. Incorrect. First part is incorrect. Plausible because 2 out of 3 ASO bi-stables tripped is a demand for a turbine trip. Incorrect in that ASO indications is not a direct indication of a turbine trip. Second part is incorrect. Plausible because stopping EHC pumps is possible from the control room. Incorrect in that FRSM-1 step 6 directs locally tripping the turbine at the front standard.

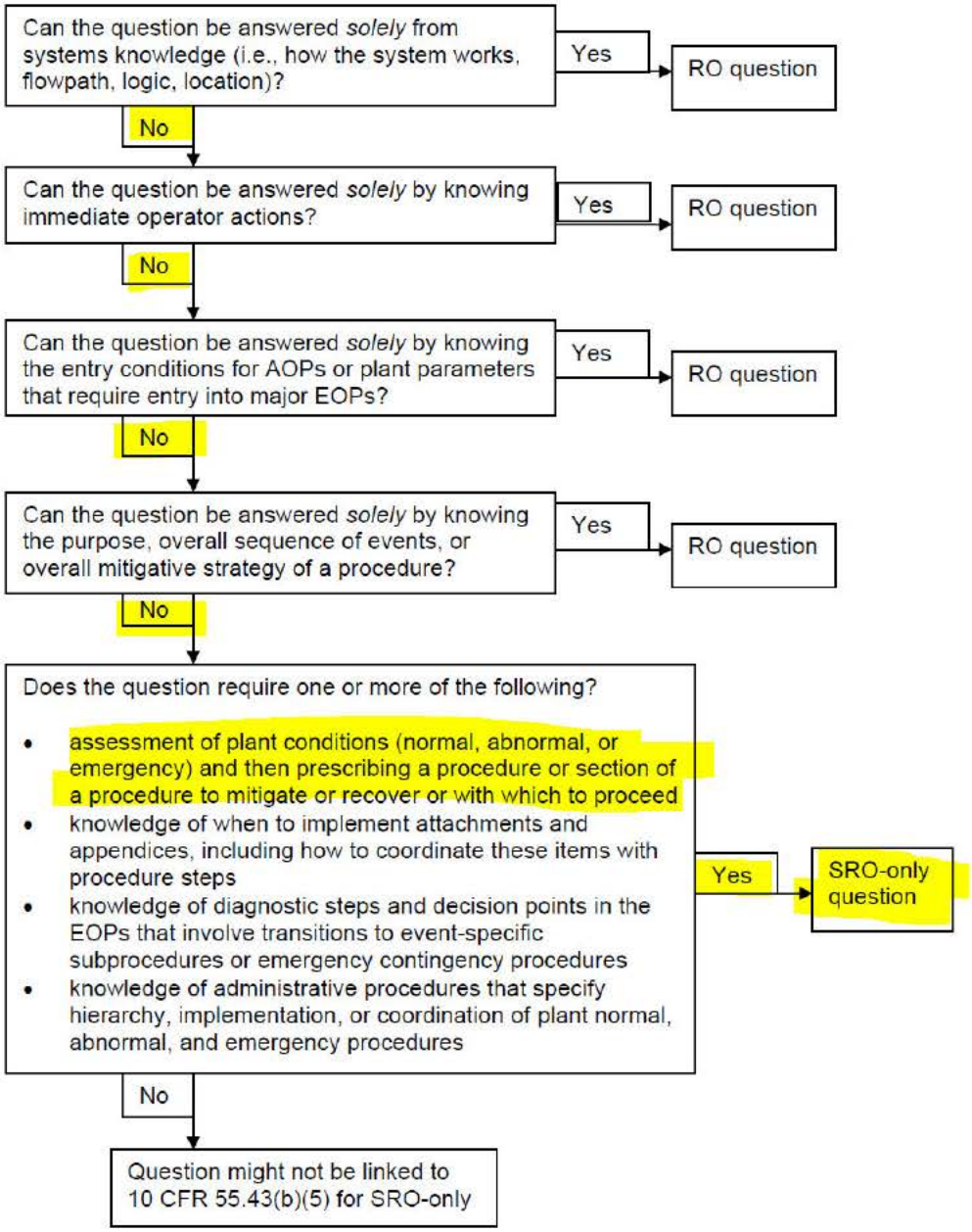
D. Incorrect. First part is incorrect. Plausible because 2 out of 3 ASO bi-stables tripped is a demand for a turbine trip. Incorrect in that ASO indications is not a direct indication of a turbine trip. Second part is correct.

Technical References:	2-EOP-FRSM-1 (R43)
Proposed References to be provided:	None
Learning Objective:	NOS05FRSM00-05, Objective 4.iii. Describe the plant response to actions taken in the following EOP step sequence(s): Turbine Trip Verification.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	SRO
Category/System:	10-Anticipated Transient Without Scram
K/A #:	EPE 29 EA2.09 Ability to determine and/or interpret the following as they apply to ATWS: Occurrence of a main turbine trip.
Importance:	4.1
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the candidate is being asked how to determine if the main turbine is tripped per EOP-FRSM-1 and what action is required if the turbine fails to trip.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



81 **11-197** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- The crew is performing S2.OP-AB.SG-0001, Steam Generator Tube Leak, due to a confirmed tube leak on one SG
- Chemistry is performing hourly trends of primary to secondary leak rates

Chemistry leak rates are as follows:

Time	SG Leak Rate (gpd)
0200	40
0300	60
0400	90
0500	125

Based on the above trends, what is the HIGHEST action that the CRS should direct in accordance with S2.OP-AB.SG-0001?

[REFERENCE PROVIDED]

A. Be in MODE 3 in less than or equal to 24 hours.

B. Be in MODE 3 within 6 hours.

C. Reduce power to less than or equal to 50% within 1 hour AND be in MODE 3 in the following 2 hours.

D. Trip the reactor and actuate Safety Injection.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because this is an action per Attachment 1 for Action Level 1.

B. Incorrect. Plausible because this is an action per Attachment 1 for Action Level 3 if leakrate is ≥ 150 gpd.

C. Correct. Per S2.OP-AB.SG-0001 Attachment 1 step 1.B (Action Level 3) with leak rate at greater than or equal to 75 gpd and a rate of change greater than or equal to 30 gpd/hr, then reduce power less than or equal to 50% in 1 hour and place the unit in MODE 3 in the following 2 hours.

D. Incorrect. Plausible because this is an action directed in Attachment 1.

Technical References:	S2.OP-AB.SG-0001 (R34)
Proposed References to be provided:	**Attachment 1 from S2.OP-AB.SG-0001**
Learning Objective:	NOS05ABSGTL-06, Objective 3, Given a set of initial plant conditions: Describe the plant response to actions taken in the abnormal procedure
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.12
Level:	SRO
Category/System:	11-Steam Generator Tube leak / Conduct of Operations
K/A #:	APE 037 G2.1.25 Ability to interpret reference materials, such as graphs, curves, and tables (reference potential).
Importance:	4.2
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the SRO is required to interpret plant data and determine what action is required based on SG tube leak rate and rate of change per S2.OP-AB.SG-0001.

Additional Reference:

S2.OP-AB.SG-0001(Q)

ATTACHMENT 1, CONTINUOUS ACTION SUMMARY
(Page 4 of 7)

NOTE

___ ♦ Unit shutdown is required when the following action levels are exceeded. Chemistry is to obtain grab samples to confirm the leak rate during the unit shutdown.

___ ♦ It is mandatory that plants without a second radiation monitor shutdown based on the indications of one monitor.

___ ♦ Unit shutdown is **NOT** to be delayed while waiting for sample results. Shutdown actions may only be suspended when sample results confirm that total Primary to Secondary leakage is below the unit shutdown criteria as listed in Step 1.7.

___ ♦ 2R53 data is valid only in Mode 1, N-16 is only generated in sufficient amounts greater than or equal to 25% power.

___ 1.7 Action Levels 2 and 3, shutdown required

___ A. Action Level 2,

1. **IF AT ANY TIME** the leak rate is quantitatively confirmed at greater than or equal to 75 gpd in ANY SG and sustained for greater than or equal to 1 hr but at a rate of Less than or equal to 30 gpd/hr,

___ **THEN PERFORM** the following:

___ ♦ **PLACE** Unit 2 in Mode 3 in Less than or equal to 24 hr

___ ♦ **MONITOR** rate of increase in leak rate

___ ♦ **IDENTIFY** leaking SG and quantify leakage

___ ♦ **CONTAIN** systems to minimize spread of contamination

___ ♦ **INCREASE** frequency of available RMS monitoring

Additional Reference:

S2.OP-AB.SG-0001(Q)

ATTACHMENT 1, CONTINUOUS ACTION SUMMARY
(Page 5 of 7)

1.7 (continued)

B. Action Level 3, [CRCA 70148825]

1. **IF AT ANY TIME** the leak rate is quantitatively confirmed at greater than or equal to 75 gpd and a rate of change greater than or equal to 30 gpd/hr

   OR the leak rate is quantitatively confirmed at greater than or equal to 75 gpd and the continuous radiation monitors are unavailable, (both R53 and R19 monitors are unavailable)

___ **THEN PERFORM** the following:

___ ♦ **REDUCE** power to less than or equal to 50% in 1 hr IAW S2.OP-AB.LOAD-0001(Q) Rapid Power Reduction

___ ♦ **PLACE** Unit 2 in Mode 3 in following 2 hrs

___ ♦ **IDENTIFY** leaking SG and quantify leakage

___ ♦ **CONTAIN** systems to minimize spread of contamination

___ ♦ **INCREASE** frequency of available RMS monitoring

2. **IF AT ANY TIME** the leak rate is quantitatively confirmed at greater than or equal to 150 gpd,

___ **THEN PERFORM** the following:

___ ♦ **PLACE** Unit 2 in Mode 3 in less than or equal to 6 hrs

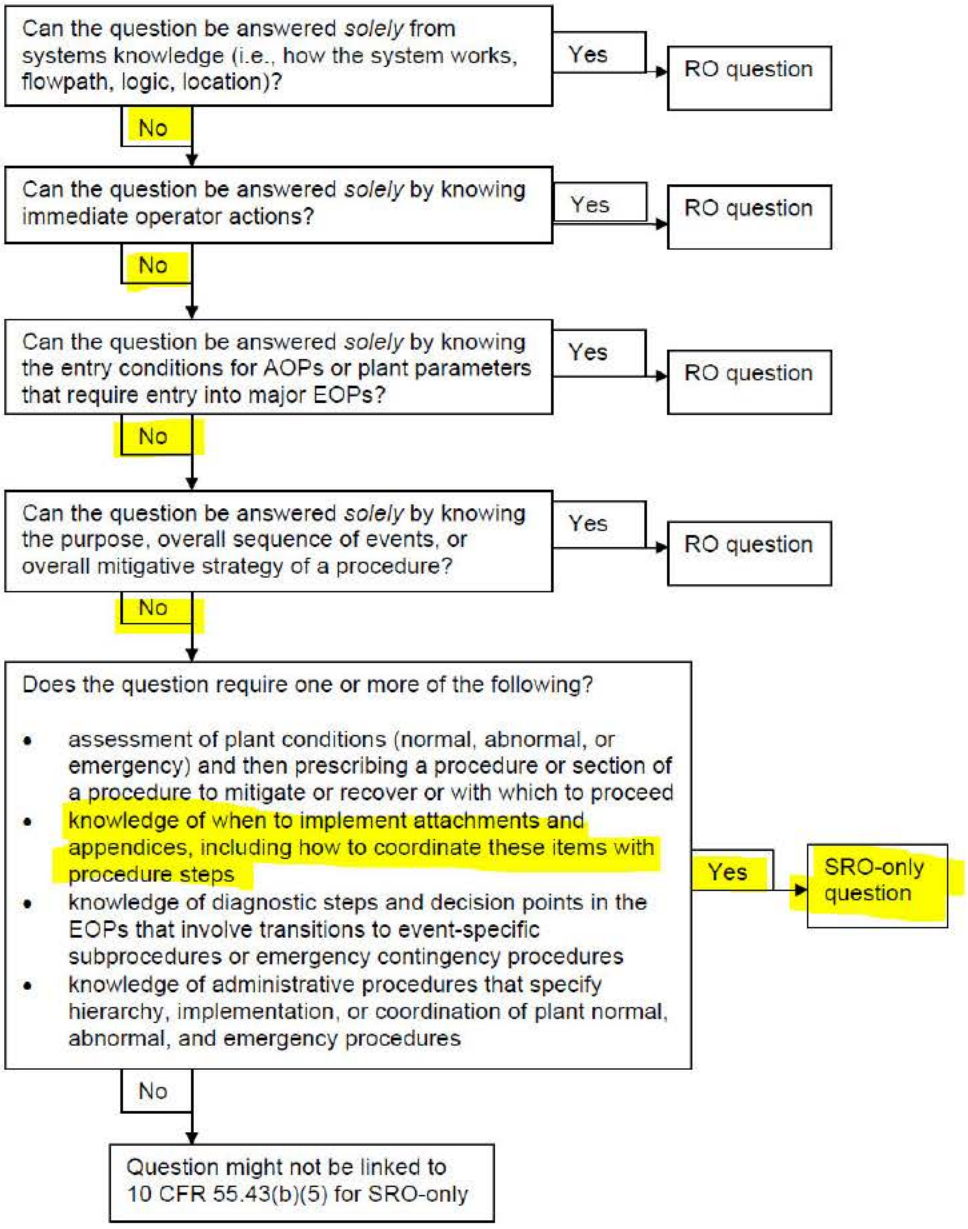
___ ♦ **IDENTIFY** leaking SG and quantify leakage

___ ♦ **CONTAIN** systems to minimize spread of contamination

___ ♦ **INCREASE** frequency of available RMS monitoring

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



82 **11-198** **Points: 1.00**

Given:

- Salem Unit 2 is performing a reactor startup
- A valid P-6 permissive light is now LIT on 2RP4

When the P-6 permissive was LIT, the RO observed the following:

- 2N35, Intermediate Range, NIS channel indicates 1E-5% power
- 2N36, Intermediate Range, NIS channel indicates 1E-8% power

Which of the following completes both statements?

1. The __(1)__ Intermediate Range (IR) NIS channel is not indicating as expected and is INOPERABLE.

2. Per TS 3.3.1.1 Instrumentation Bases, why is the 24 hour action time allowed when one IR NIS channel is INOPERABLE?

A. (1) 2N35
(2) Takes into account the low probability of failure of the operable IR channel during this period.

B. (1) 2N35
(2) Takes into account the low probability of reactivity events during this period.

C. (1) 2N36
(2) Takes into account the low probability of reactivity events during this period.

D. (1) 2N36
(2) Takes into account the low probability of failure of the operable IR channel during this period.

Answer: D

Explanation / Justification:

A. Incorrect. Part 1 is incorrect. Plausible because the operator can incorrectly recall the point where the SR and IR overlap when P-6 is present. Part 2 is correct.

B. Incorrect. Part 1 is incorrect. Plausible because the operator can incorrectly recall the point where the SR and IR overlap when P-6 is present. Part 2 is incorrect. Plausible because the IR channel does provide redundant trip function to the PR NIS low setting trip, but this is not the bases for the 24 hour action limit.

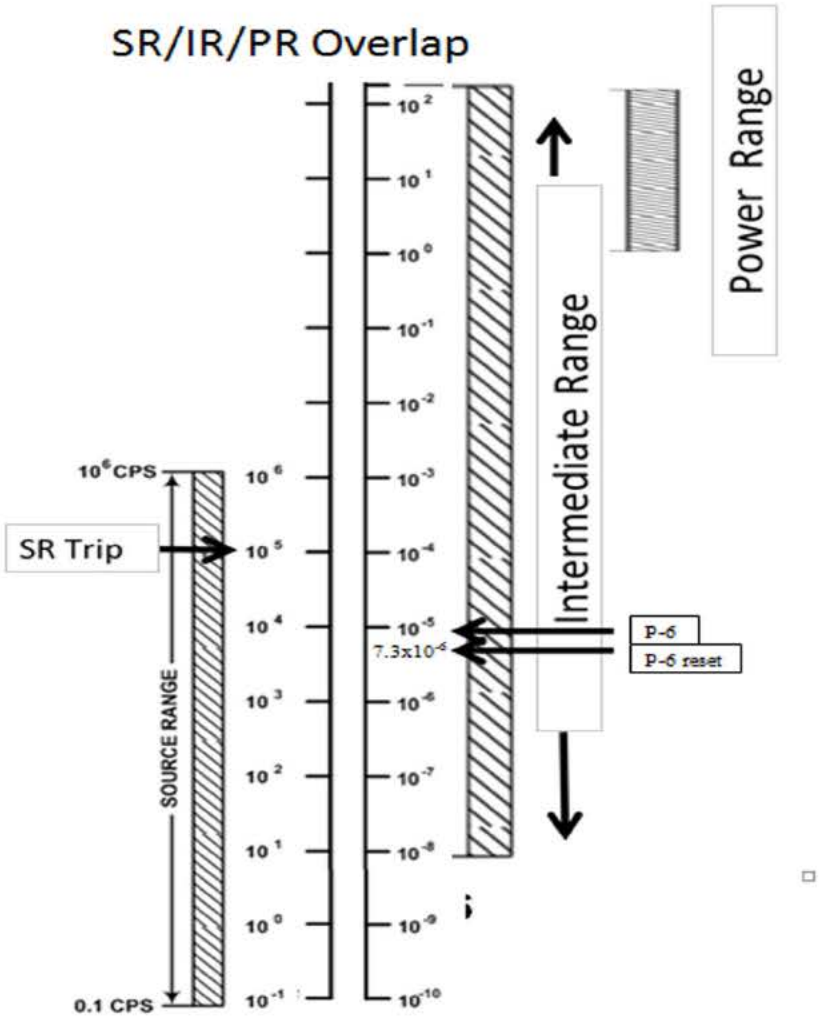
C. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because the IR channel does provide redundant trip function to the PR NIS low setting trip, but this is not the bases for the 24 hour action limit.

D. Correct. When P-6 is LIT, the IR NIS would indicate 1E-5% power. Therefore, 2N36 IR channel is not indicating for the expected overlap between the SR channels. Per TS 3.3.1.1 bases, the IR NIS channel high flux trip ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup. The trip function provides redundant protection to the Power Range High Flux – Low Setpoint trip function. The 24 hour action times for one inoperable IR channel allow for a slow and controlled adjustment above P-10 or below P-6 and take into account the redundant capability afforded by the operable IR channel and its low probability of failure during this period.

Technical References:	TS 3.3.1.1 bases
Proposed References to be provided:	None
Learning Objective:	NOS05ABNIS1-04, Objective 2. Describe, in general terms, the actions taken in S2.OP-AB.NIS-0001(Q) and their Bases, in accordance with this lesson plan
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	SRO
Category/System:	11-Loss of Intermediate Range Nuclear Instrumentation
K/A #:	APE 33 AA2.01 Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Equivalency among source range, intermediate range, and power range channel readings.
Importance:	3.4
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

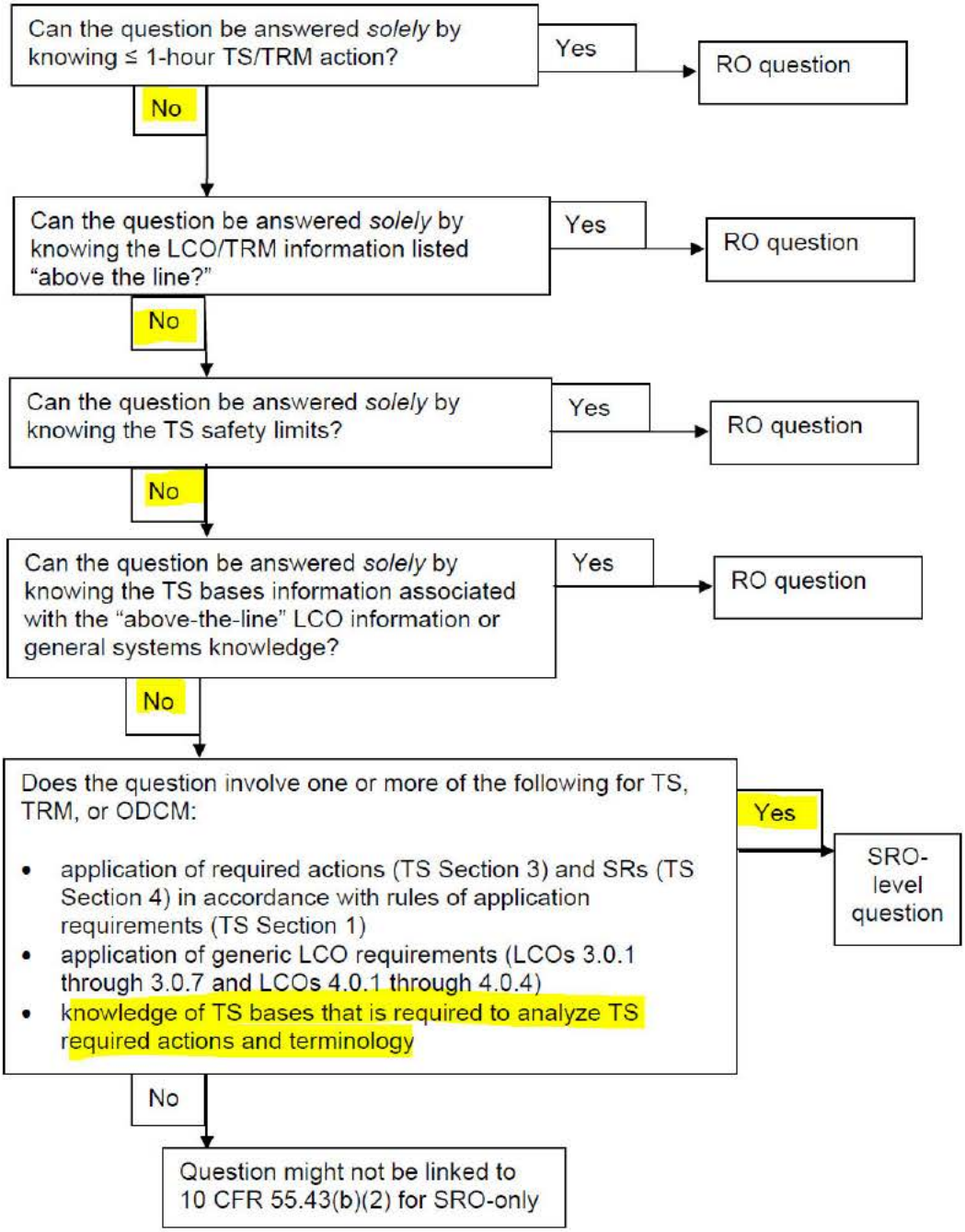
K/A Match: The K/A is matched because the candidate is being asked what the equivalency reading would be between the SR and IR NIS when the P-6 bistable is LIT and what is the TS bases for the 24 hour action time.

Additional Reference:



SRO Match Justification:

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



83 **11-199** **Points: 1.00**

Given:

- BOTH Salem Control Rooms have been evacuated due to toxic gas

Per S2.OP-AB.CR-0001, Control Room Evacuation, which of the following completes both statements?

1. EOPs __(1)__ applicable during Control Room Evacuation.
2. The "Caution" statement in Attachment 3 alerts the CRS to monitor SG pressures at the Hot Shutdown Panel to avoid __(2)__.

A. (1) are NOT
(2) exceeding Tech Spec RCS cooldown rate of 100 °F per hour

B. (1) are NOT
(2) Safety Injection on steam line differential pressure

C. (1) are
(2) exceeding Tech Spec RCS cooldown rate of 100 °F per hour

D. (1) are
(2) Safety Injection on steam line differential pressure

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the SRO may that TS limit of 100 ° F per hour still apply in this procedure. Incorrect in that the RCS cooldown limit is 25 °F per hour IAW Attachment 3 step 24.0.

B. Correct. Per the Note in section 2.0 Immediate Actions in S2.OP-AB.CR-0001, EOPs are NOT applicable during Control Room Evacuation. EOPs should be used for information only or as directed by the TSC while performing this procedure. The CRS will monitor SG pressures at the Hot Shutdown Panel IAW Attachment 3 step 5.0. A Caution statement in step 5.0 cautions the CRS that a steam line differential pressure SI may occur if DP exceeds 100 psi differential pressure

C. Incorrect. Plausible because EOP-TRIP-1 is the typical procedure to enter following a reactor trip. Plausible because the SRO may that TS limit of 100 ° F per hour still apply in this procedure. Incorrect in that the RCS cooldown limit is 25 °F per hour IAW Attachment 3 step 24.0.

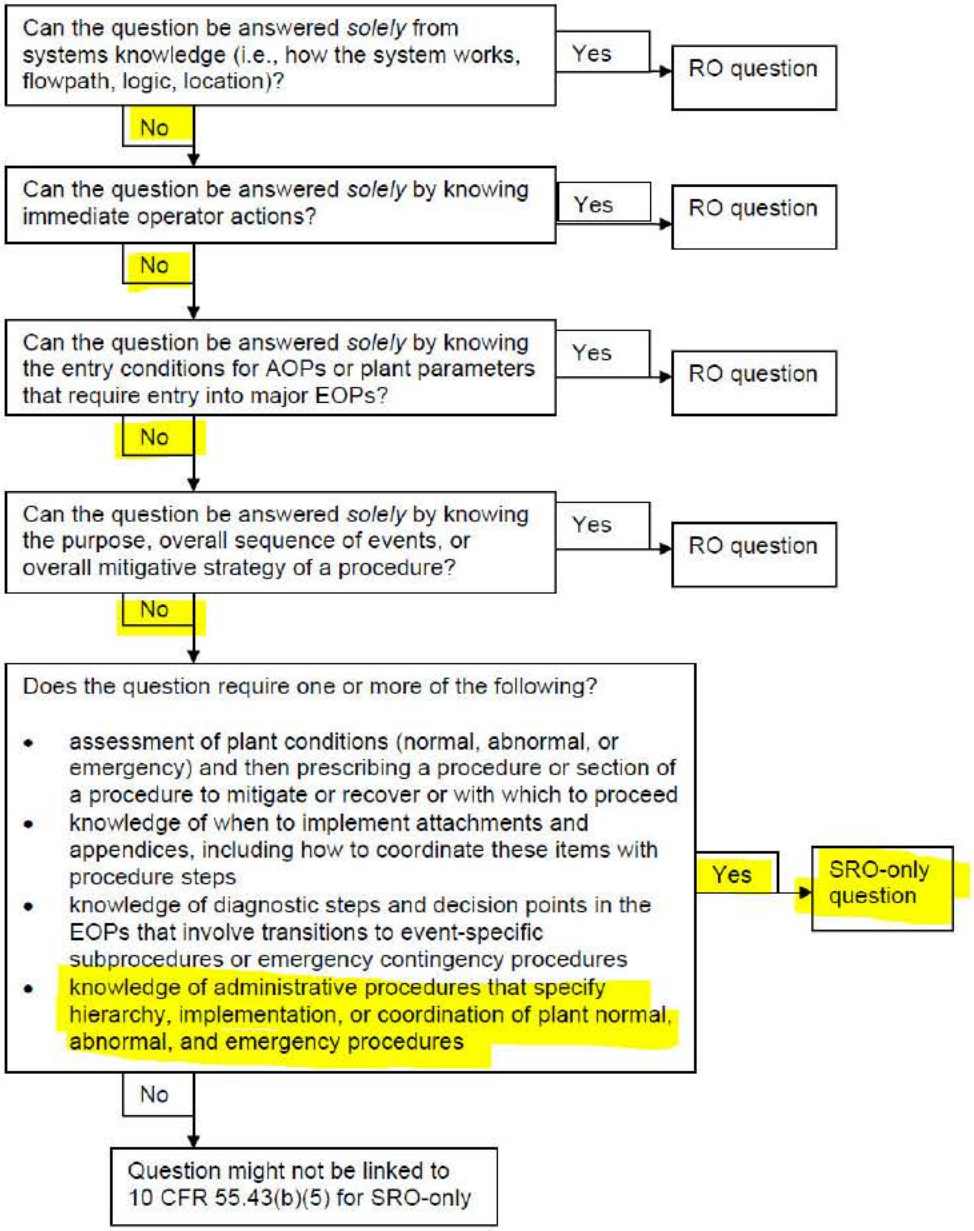
D. Incorrect. Plausible because EOP-TRIP-1 is the typical procedure to enter following a reactor trip.

Technical References:	S2.OP-AB.CR-0001 (R24)
Proposed References to be provided:	None
Learning Objective:	NOS05ABCR01-06 Objective 2. Describe, in general terms, the actions taken in S1/S2.OP-AB.CR-0001 and the bases for the actions.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	SRO
Category/System:	11-Control Room Evacuation / Emergency Procedures/Plan
K/A #:	APE 68 G2.4.20 Knowledge of the operational implications of emergency and abnormal operating procedures warnings, cautions, and notes.
Importance:	4.3
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the question is asking the SRO knowledge of a note in the Control Room Evacuation abnormal procedure and how to apply the note.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



84 **10-147** **Points: 1.00**

Given:

- Salem Unit 2 is performing 2-EOP-FRCC-1, Response to Inadequate Core Cooling
- All steam generators are intact and at approximately 1040 psig
- MSIVs are open
- The crew is preparing to depressurize the steam generators in order to inject accumulators
- Containment Pressure is 5 psig

Which of the following completes both statements?

1. The operators will depressurize the steam generators using steam dumps at a __(1)__.
2. Following SG depressurization, the SG NR levels are as follows:

21 SG	22 SG	23 SG	24 SG
7%	12%	16%	10%

Per 2-EOP-FRCC-1, step 18, RCP Start for Temporary Core Cooling, what is the MAXIMUM number of RCPs that the CRS will direct to be started (assume CETs remain above 1200 F)?

A. (1) maximum rate while attempting to avoid a Main Steam Isolation
(2) One

B. (1) maximum rate while attempting to avoid a Main Steam Isolation
(2) Three

C. (1) rate not to exceed 100 °F per hour
(2) Three

D. (1) rate not to exceed 100 °F per hour
(2) One

Answer: A

Explanation / Justification:

A. Correct. Part 1 is correct. Part 2 is correct. Per 2-EOP-FRCC-1 step 11, the crew will depressurize the intact SGs by using the steam dumps or MS10s at a maximum rate. At step 18 with CETs above 1200 °F, then the crew will determine how many RCPs can be started to reduce CETs. With containment at adverse conditions, the required NR level to start RCPs will be 15%, therefore, after performing step 18, only one RCP meets the requirements in step 18 to be started.

B. Incorrect. Plausible because if use normal containment conditions, the required NR level is 9% and then three RCPs would be started.

C. Incorrect. Part 1 is incorrect. Plausible because this RCS cooldown rate is used on other EOPs to avoid violating tech spec limits. Part 2 is incorrect. Plausible because if use normal containment conditions, the required NR level is 9% and then three RCPs would be started.

D. Incorrect. Part 1 is incorrect. Plausible because this RCS cooldown rate is used on other EOPs to avoid violating tech spec limits. Part 2 is correct.

Technical References:	2-EOP-FRCC-1 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05FRCC00-06 Objective 2.a. Describe the EOP mitigation strategy for the following: Response to Inadequate Core Cooling
Question Source:	Modified Bank – Diablo Canyon 2019 NRC SRO84
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	SRO
Category/System:	10-Inadequate Core Cooling
K/A #:	EPE 74 EA2.12 Ability to determine and/or interpret the following as they apply to the Inadequate Core Cooling: RCS cooldown rate
Importance:	3.8
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the SRO must determine the RCS cooldown rate during depressurizing SGs to inject the accumulators and select the procedure to enter if all attempts to restore core cooling fails in order to mitigate inadequate core cooling.

Parent Question: Diablo Canyon 2019 NRC SRO84

Modified Justification: Part 1 of this questions was changed to test the SRO on the cooldown rate in 2-EOP-FRCC-1.

Question 84

GIVEN

- The crew is performing EOP FR-C.1, Response to Inadequate Core Cooling
- All RCPs are available
- Containment pressure is 1.1 psig and stable
- Narrow Range Steam Generator Levels are:
  - 1-1 ~ 24%
  - 1-2 ~ 12%
  - 1-3 ~ 18%
  - 1-4 ~ 0%

1) In accordance with step 19, Restore Temporary Core Cooling, of EOP FR-C.1, what is the MAXIMUM number of RCS cooling loops available?

2) If, after the RCPs are started in the available RCS cooling loops and all PORVs and head vents are open, core exit thermocouples are greater than 1200°F and rising, the Shift Foreman should:

A. 1) two
   2) GO TO SACRG-1, Severe Accident Control Room Guideline Initial Response

B. 1) two
   2) return to step 1, Ensure ECCS Valve Alignment – Proper Emergency Alignment

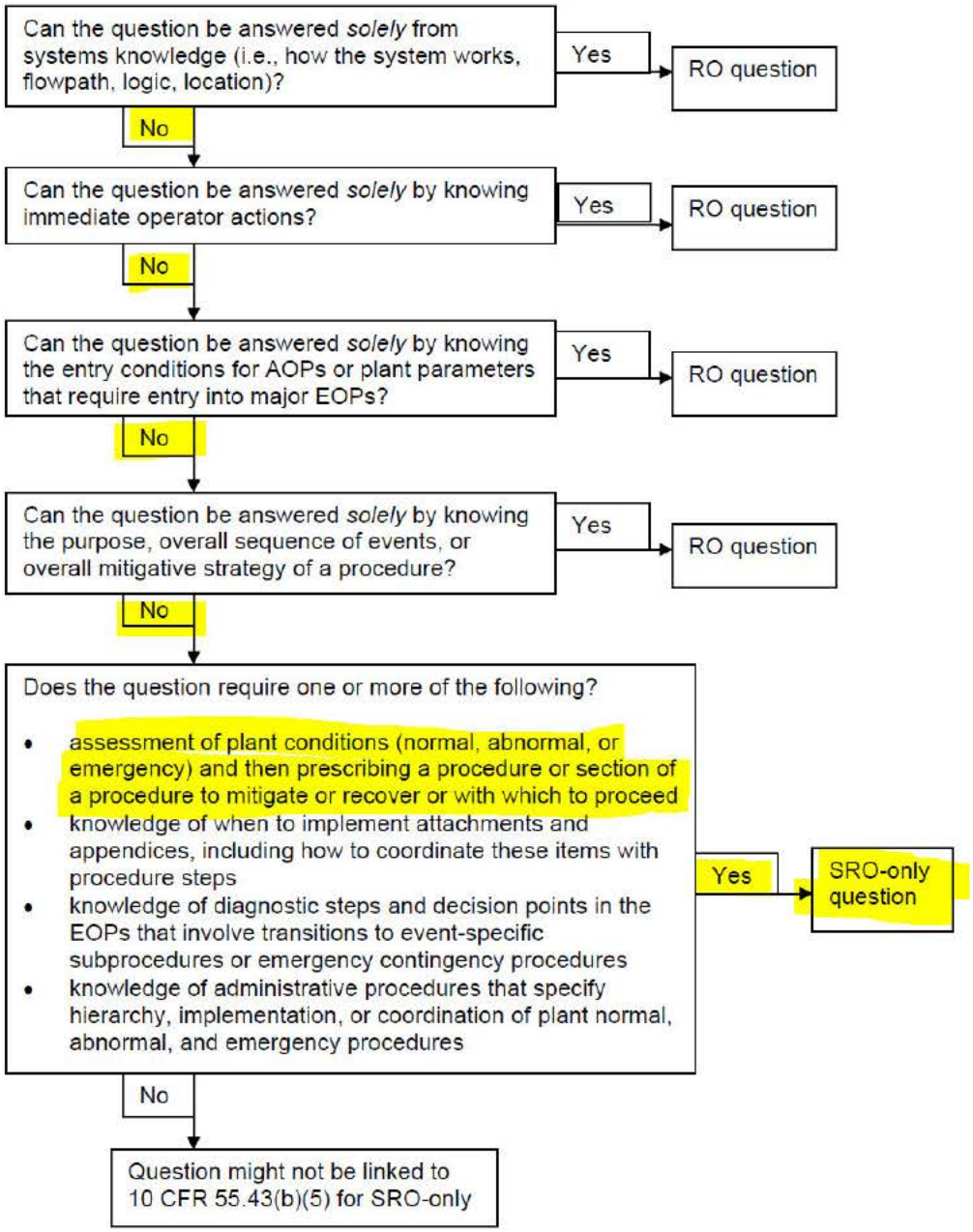
C. 1) three
   2) GO TO SACRG-1, Severe Accident Control Room Guideline Initial Response

D. 1) three
   2) return to step 1, Ensure ECCS Valve Alignment – Proper Emergency Alignment

Proposed Answer: A. 1) two
2) GO TO SACRG-1, Severe Accident Control Room Guideline Initial Response

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



85 **10-148** **Points: 1.00**

Given:

Redacted Due to Security-Related Information

Redacted Due to Security-Related Information

Answer: A

Redacted Due to Security-Related Information

Explanation / Justification:

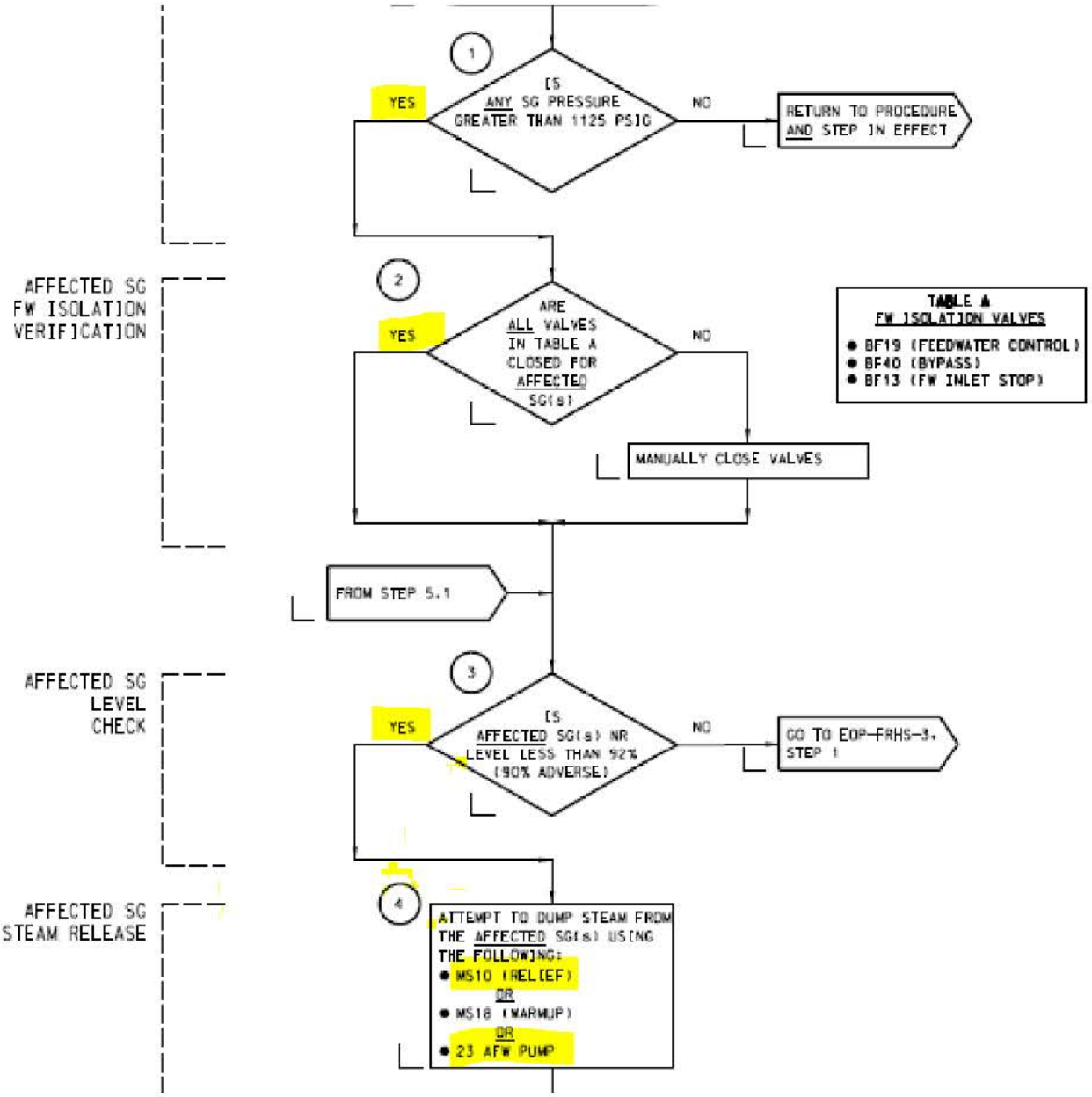
A. [redacted]

Technical References:	2-EOP-FRHS-2 (R41), [redacted]
Proposed References to be provided:	None
Learning Objective:	NOS05ABCR04-03, Objective 1. Describe, in general terms, the actions taken in [redacted] and the bases for the actions .
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	SRO
Category/System:	10-Steam Generator Overpressure / Emergency Procedures/Plan
K/A #:	W E13 G2.4.28 Knowledge of the procedures relating to a security event (ensure that the test item includes no safeguards information).
Importance:	4.1
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the SRO is being asked knowledge of site security procedures and how to implement them during a SG overpressure event.

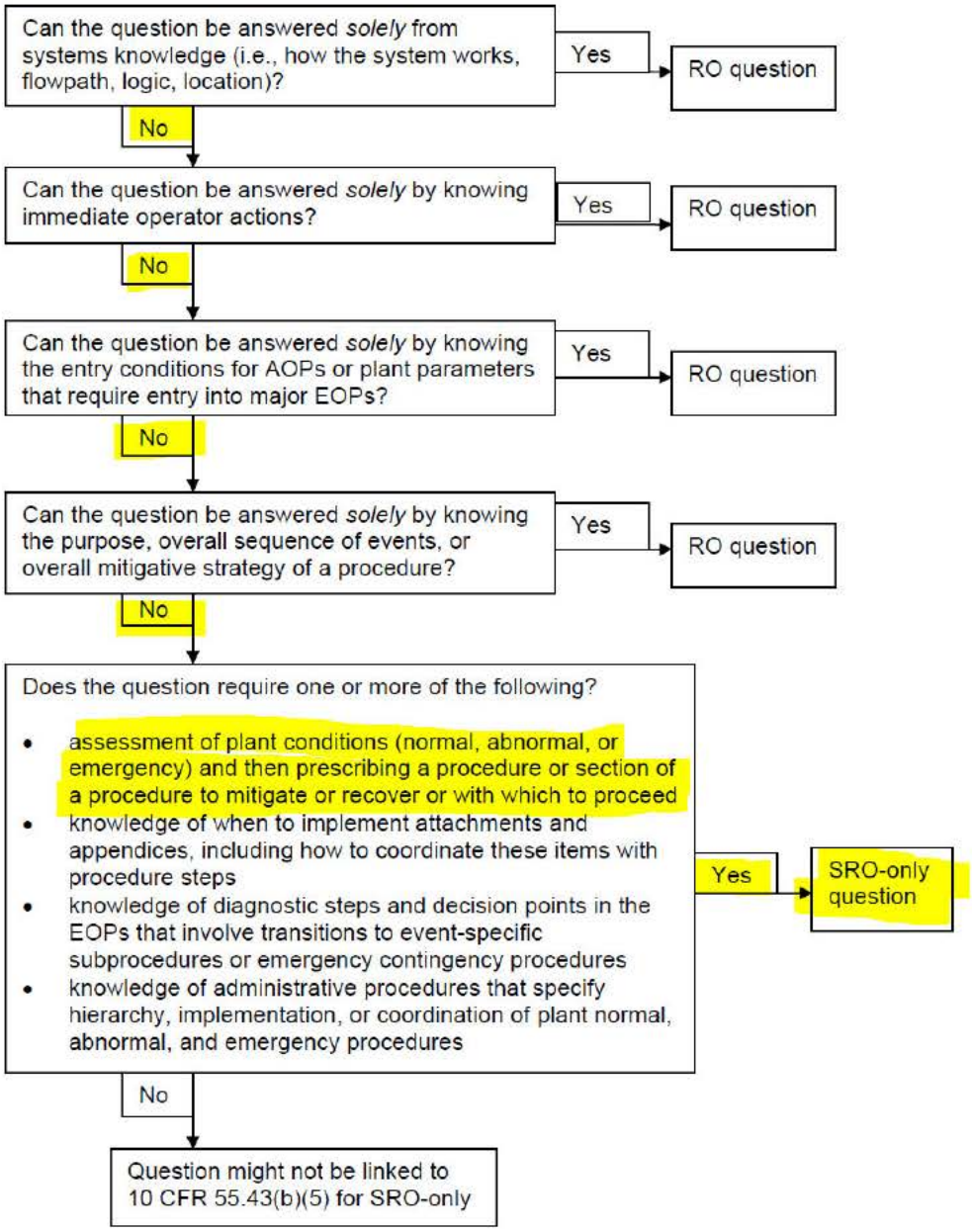
Redacted Due to Security-Related Information

Additional References: 2-EOP-FRHS-2



SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



86 **04-071** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

Then:

- Console Alarm, STANDPIPE LEVEL-LO, is now LIT
- The RO reports the following for 21 RCP:
  - Seal Leak-off recorder indicates 1.5 gpm and steady
  - Seal Injection Flow indicates 8.5 gpm

Which of the following completes both statements?

1. Which RCP seal failed?
2. What action(s) will the CRS direct the crew?

Note: ARP = Alarm Response Procedure

S2.OP-AB.RCP-0001, Reactor Coolant Pump Abnormality

A. 1) #3 seal
2) Perform an orderly shutdown and stop 21 RCP within 8 hours per S2.OP-AB.RCP-0001.

B. 1) #3 seal
2) Make-up to the standpipe per the ARP; power operation may continue.

C. 1) #2 seal
2) Make-up to the standpipe per the ARP; power operation may continue.

D. 1) #2 seal
2) Perform an orderly shutdown and stop 21 RCP within 8 hours per S2.OP-AB.RCP-0001.

Answer: B

Explanation / Justification:

A. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because this is an action in S2.OP-AB.RCP-0001 for a leaking or degraded #2 seal.

B. Correct. The conditions in the stem describes a #3 seal leak. Standpipe level low is a primary indicator that the #3 seal is leaking only since the #1 seal leakoff flow is steady at 1.5 gpm which is normal conditions for an intact #2 seal. The crew will take action per the console ARP and perform a makeup to the clear the low standpipe level alarm and initiate S2.OP-ST.RC-0008 for RCS leak rate. The CRS could enter S2.OP-AB.RCP-0001 based on the console alarm, but the AB will not provide any direction for a leaking #3 seal.

C. Incorrect. Part 1 is incorrect. Plausible because the operator may confuse the standpipe level indications in the stem as signs of a #2 seal leak. Part 2 is correct.

D. Incorrect. Part 1 is incorrect. Plausible because the operator may confuse the standpipe level indications in the stem as signs of a #2 seal leak. Part 2 is incorrect. Plausible because this is an action in S2.OP-AB.RCP-0001 for a leaking or degraded #2 seal.

Technical References:	S2.OP-AB.RCP-0001 (R28)
Proposed References to be provided:	N/A
Learning Objective:	NOS05ABRCPUMP-18, Objective 17.b. Given any of the following and appropriate control room reference material, evaluate and determine the effect on the Reactor Coolant Pump: Degraded Reactor Coolant Pump seals and seal water
Question Source:	Modified Bank – Diablo Canyon 2019 NRC SRO86
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.6
Level:	SRO
Category/System:	04-Reactor Coolant Pump System
K/A #:	003 A2.01 Ability to (a) predict the impacts of the following on the Reactor Coolant Pump System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Problems with RCP seals, especially seal leakoff rates
Importance:	4.0
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked to predict the impact of a low RCP standpipe level and based on plant conditions to determine the course of action based on the failed or degraded #3 RCP seal.

Parent Question: Diablo Canyon 2019 SRO86

Modified question to include Part 2 to ask knowledge of what specific action in the AB is required based on the type of seal failure.

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	2
003 A2.01 - Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems with RCP seals, especially rates of seal leak-off	**Group #**	1
	K/A #	003 A2.01
	Rating	3.9

Question 86

Unit 1 is at 100% power.

The following events occur:

- PK 05-01, RCP No.11, alarms, due to input 1259, RCP 1-1 No. 2 Seal Leakoff Flow High
- RCP 1-1 Number 1 seal leakoff is approximately 1.0 gpm
- RCP 1-2, 1-3, 1-4 Number 1 seal leakoff is approximately 3.0 gpm per RCP
- Aux watch reports an increase in RCDT fill rate
- Charging flow is approximately 87 gpm

In accordance with AR PK05-01, what action should be taken by the Shift Foreman?

A. Direct the operator to raise seal injection flow.

B. Direct the operator to trip the reactor, stop RCP 1-1 and enter E-0, Reactor Trip or Safety Injection.

C. Go to OP AP-28, RCP Malfunction, Section B, "RCP Number 1 Seal Failure".

D. Go to OP AP-28, RCP Malfunction, Section C, "RCP Number 2 or 3 Seal Failure".

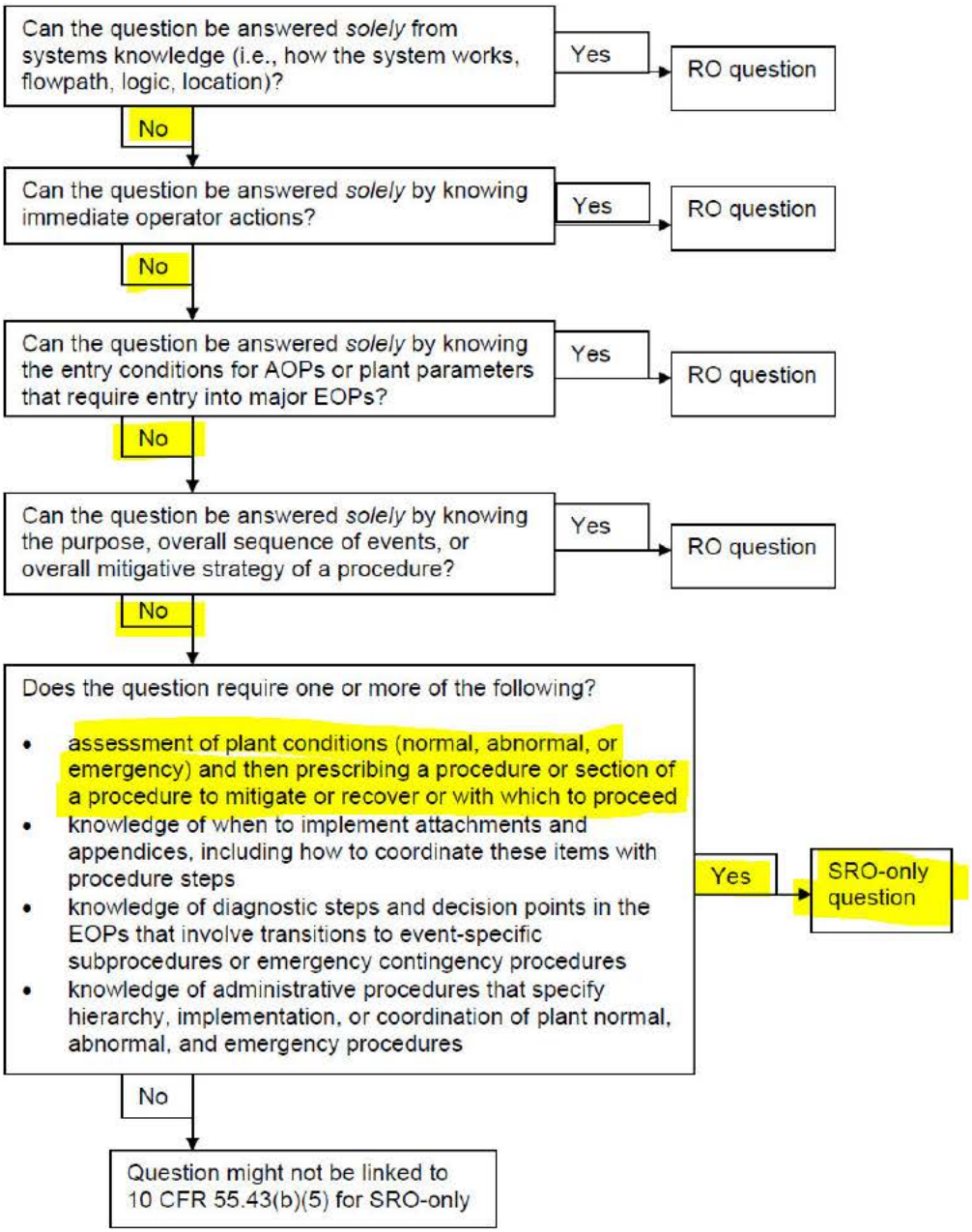
Proposed Answer: D. Go to OP AP-28, RCP Malfunction, Section C, "RCP Number 2 or 3 Seal Failure".

Explanation:

A. Incorrect. Injection flow is lower, but raising it will not address the failure.
B. Incorrect. This is the action in the PK if there are multiple diverse alarms.
C. Incorrect. the 1259 Input directs going to step 2.12 in the AR PK. This step, has actions for both a #1 and #2 or #3 seal failure.This is the action at this step in AR PK for a number 1 seal failure. Indications: Rise in pump radial bearing AND seal outlet temperatures, • Drop in Number 1 Seal DP, • Rise in Number 1 Seal leakoff flow, • Rise in Number 2 Seal leakoff flow, • Rise in Seal injection flow
D. Correct. Lower Number 1 Seal leakoff flow corresponding to the rise in Number 2 Seal leakoff flow, • More frequent filling of Number 3 Seal Standpipe, • Rise in pump vibration, • Recent rise in Containment Sump fill rate

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



87 **10-149** **Points: 1.00**

Given:

- A Loss of Offsite Power has occurred on Salem Unit 2
- The crew is performing actions in 2-EOP-TRIP-4, Natural Circulation Cooldown
- 21 Charging Pump is running
- RCS cooldown to establish required subcooling is in progress
- Low PZR Pressure SI signal is BLOCKED
- RCS temperature is 537 °F

Then:

- RCS pressure is 1183 psig and lowering
- 2CV55, Charging Flow Control Valve, is fully open
- 2CV71, Seal Injection Backpressure Valve, is fully open
- Pressurizer level is 10% and lowering

Which of the following describes the correct action(s) for these conditions per 2-EOP-TRIP-4?

A. Stop the RCS cooldown and wait until PZR level and subcooling are restored.

B. Align the Charging Pump suction to the RWST, open the BIT isolation valves, and transition to 2-EOP-LOCA-1, Loss of Reactor or Secondary Coolant.

C. Operate Pressurizer heaters as necessary to maintain saturated conditions in the pressurizer per 2-EOP-TRIP-4.

D. Actuate Safety Injection and return to 2-EOP-TRIP-1, Reactor Trip or Safety Injection.

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because operator may believe that the reason for the PZR level lowering is due to excessive cooldown of the RCS.

B. Incorrect. Plausible because this is similar to CAS step 6.1 in EOP-TRIP-4. Incorrect in that TRIP-4 does not direct entry into EOP-LOCA-1.

C. Incorrect. Plausible because this is similar to CAS step 3 in EOP-TRIP-4. Incorrect in that PZR heaters will not energize with level < 17%.

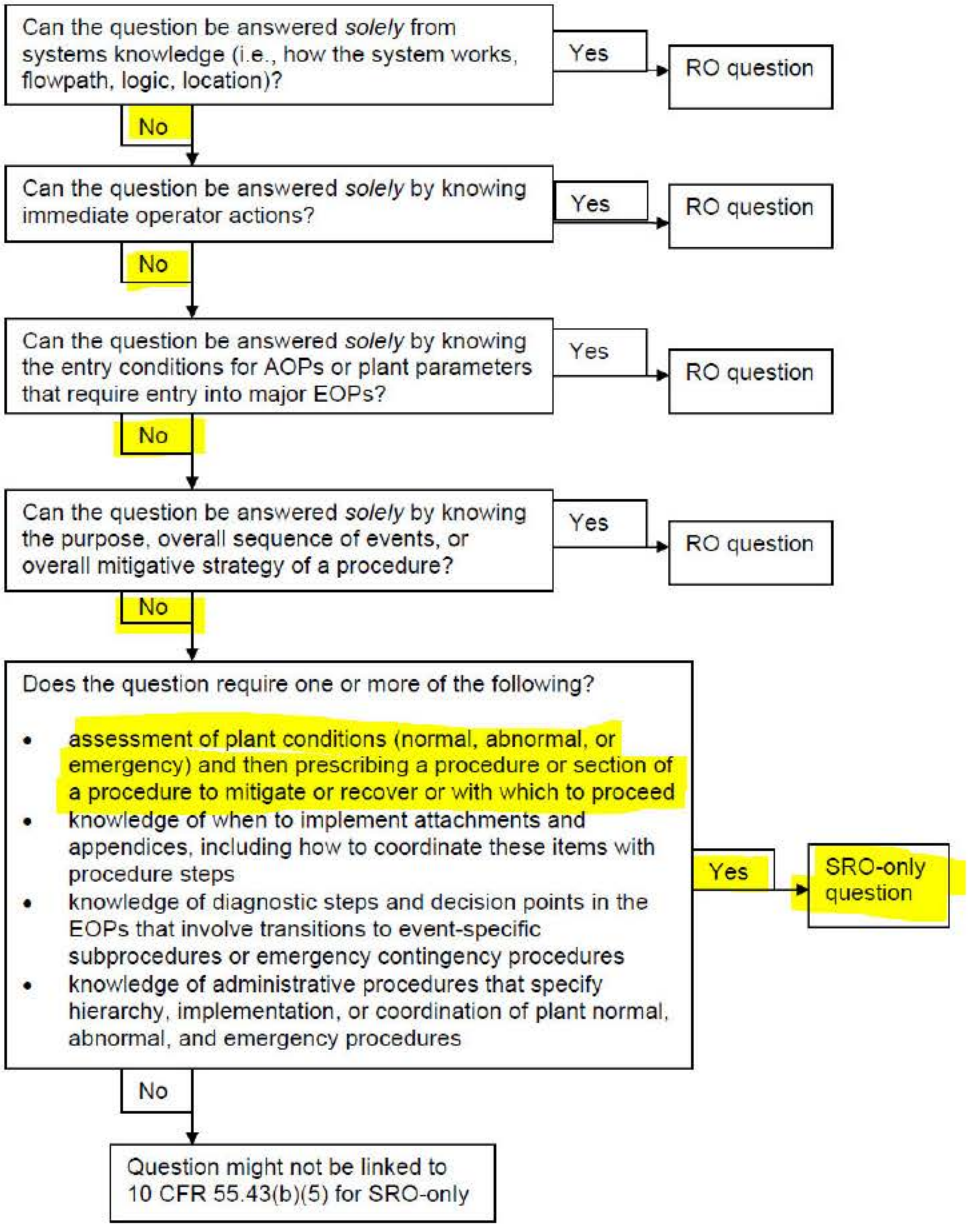
D. Correct. Per the foldout page (CAS) of 2-EOP-TRIP-4, if subcooling is 0 °F or PZR level can NOT be maintained greater than 4%, then Actuate SI and go to 2-EOP-TRIP-1. In this case subcooling is greater than 0 °F, but PZR level is lowering and cannot be maintained based on the 2CV55 and 2CV71 fully open. Letdown is isolated.

Technical References:	2-EOP-TRIP-4 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05TRP004-08, Objective 4. Determine the basis for each step, caution, note or continuous action step relative to a NATURAL CIRCULATION COOLDOWN.
Question Source:	Bank – DC Cook 2008 NRC SRO86
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.6
Level:	SRO
Category/System:	02-ECCS
K/A #:	006 A2.12 Ability to (a) predict the impacts of the following on the ECCS system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Conditions requiring actuation of ECCS.
Importance:	4.5
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked if the conditions in the stem require SI actuation of ECCS equipment.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



88 **06-027** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

Then:

- The Electric System Operator (ESO) reports a Solar Magnetic Disturbance (SMD) of greater than K-5 intensity
- Salem control room has confirmed an Excess MVAR condition

The CRS will direct the crew to…

A. trip the reactor and perform 2-EOP-TRIP-1, Reactor Trip or Safety Injection, and S2.OP-AB.GRID-0001, Abnormal Grid, concurrently.

B. trip the reactor and perform 2-EOP-TRIP-1, Reactor Trip or Safety Injection, Only.

C. initiate load reduction per S2.OP-AB.GRID-0001 Abnormal Grid, and S2.OP-AB.LOAD-0001, Rapid Load Reduction.

D. initiate load reduction per S2.OP-AB.GRID-0001 Abnormal Grid, Only.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because tripping the reactor and going to 2-EOP-TRIP-1 is directed per Attachment 1 if frequency is less than or equal to 57.55 Hz. Incorrect in that performing S2.OP-AB.GRID-0001 is not required to be performed concurrently with 2-EOP-TRIP-1

B. Incorrect. Plausible because the SRO may believe that the grid conditions in the stem require the generator to be immediately isolated from the grid for asset protection and therefore a immediate reactor trip is required.

C. Correct. Per S2.OP-AB.GRID-0001, conditions for SMD are met in step 1.3, the CRS will direct the crew perform Attachment 2, Solar Magnetic Disturbance, and initiate a load reduction to less than or equal to 77.6 % (942 MWe) at 15% per min using S2.OP-AB.LOAD-0001, Rapid Load Reduction.

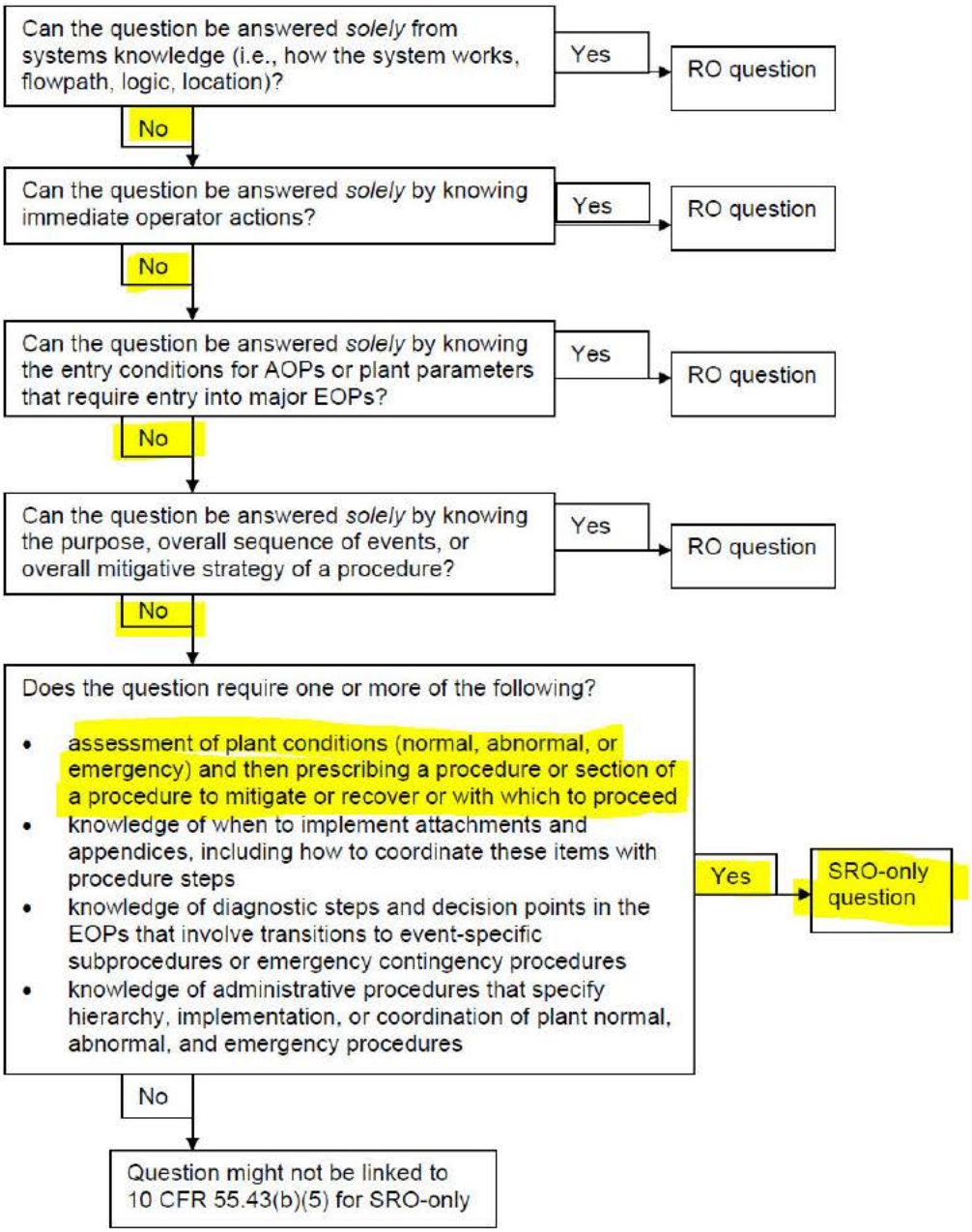
D. Incorrect. Plausible because S2.OP-AB.GRID-0001 does direct initiating a load reduction to 77.6%. Incorrect in that S2.OP-AB.GRID-0001 Attachment 2 also directs initiating the rapid load reduction per S2.OP-AB.LOAD-0001.

Technical References:	S2.OP-AB.GRID-0001 (R23)
Proposed References to be provided:	None
Learning Objective:	NOS05ABGRID-13, Objective 2. Describe, in general terms, the actions taken in S2.OP-AB.GRID-0001(Q) and the bases for the actions.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	43.5 / 45.12 / 45.13
Level:	SRO
Category/System:	06-AC Electrical Distribution System / Conduct of Operations
K/A #:	062 G2.1.6 Ability to manage the control room crew during plant transients (SRO only).
Importance:	4.8
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the question is asking the SRO how to direct the crew in the control room during a Solar Magnetic Disturbance grid transient per S2.OP-AB.GRID-0001.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



89 **06-028** **Points: 1.00**

Given:

- Both Salem Units are at 100% power.
- Severe Thunderstorms are expected to arrive onsite in 6 hours

When:

- During the monthly surveillance test on the 1A EDG, the diesel failed to start due to a failed component
- This same component is also installed in all of the other five EDGs
- The CRS enters TS 3.8.1.1 and is making preparations for performing surveillance tests on all of the five EDGs to comply with Tech Spec for common mode failure

Then:

- The Shift Manager instructs the CRSs to schedule the surveillance runs in the most expeditious manner that does NOT violate station procedures and/or commitments

In order to expeditiously complete the EDG surveillance runs prior to the arrival of the severe weather, how should the Shift Manager direct the crews on running the EDGs?

A. ONE EDG at a time on both units (i.e., 1B and 2B simultaneously).

B. ONE EDG at a time (i.e., all 5 EDGs in series).

C. ALL EDGs at the same time on one unit (i.e., 1B and 1C simultaneously, then 2A, 2B, and 2C simultaneously).

D. ALL EDGs at the same time on both units simultaneously.

Answer: A

Explanation / Justification:

A. Correct. Per S2.OP-ST.DG-0001/2/3 P&L 2.1.6, only one DG per Unit may be synchronized to the grid at a time. This prohibition minimizes any potential common failure resulting from undetected interdependence among diesel generator units (Reg Guide 1.1.08, Section C.2.b) [C0624]. Therefore, the most expeditious manner to complete the surveillance runs is to perform one EDG at a time simultaneously on each unit (i.e. 1B and 2B simultaneously).

B. Incorrect. Plausible because this method would not violate station procedures or commitments, but would not be the most expeditious method.

C. Incorrect. Plausible because the SRO may not recall the station procedure P&L or commitment on running EDGs and believe that you can run multiple EDGs on a unit.

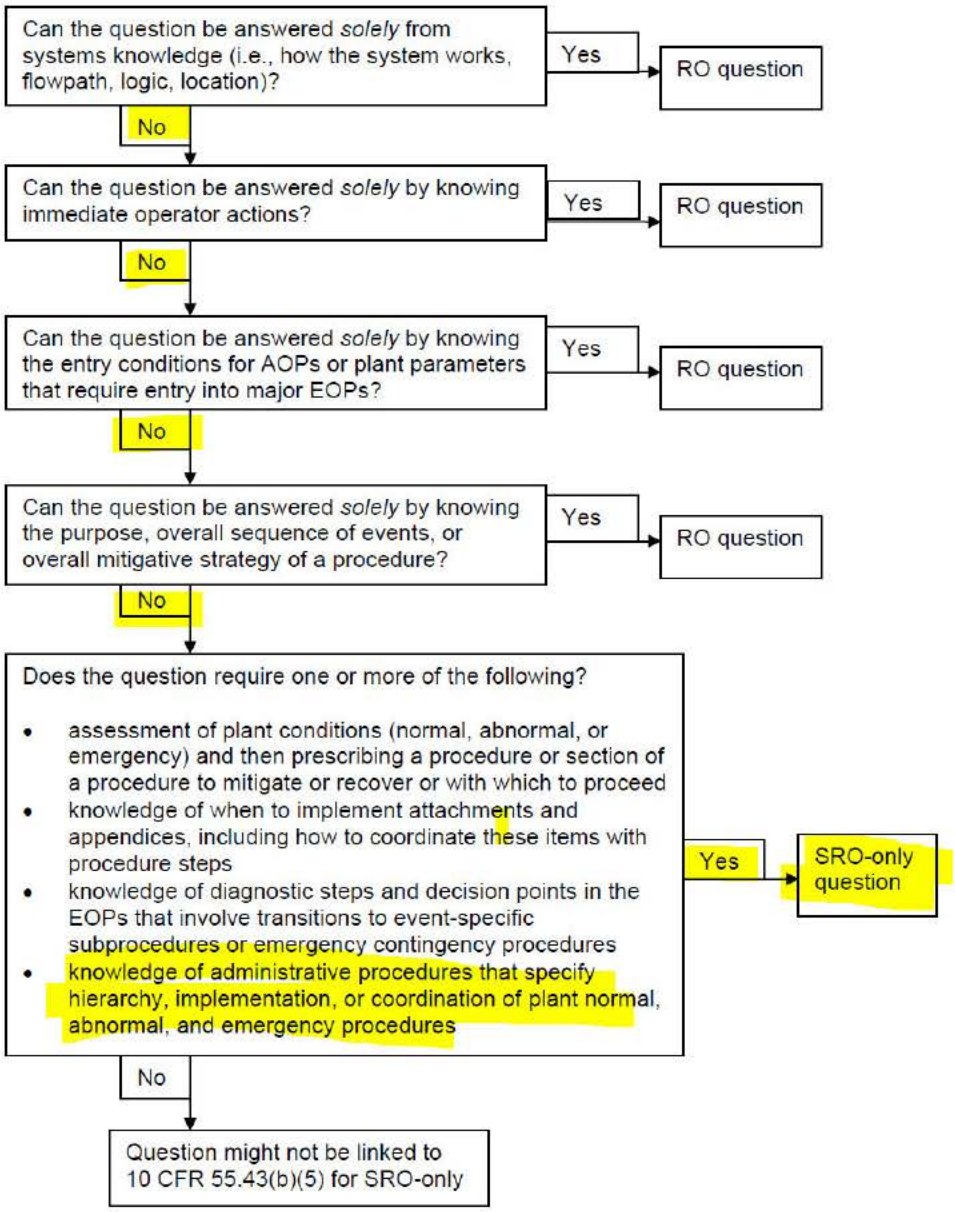
D. Incorrect. Plausible because the SRO may not recall the station procedure P&L or commitment on running EDGs and believe that you can run multiple EDGs on a unit.

Technical References:	S2.OP-ST.DG-0001 (R57)
Proposed References to be provided:	None
Learning Objective:	NOS05EDG000-15 Objective 12. Discuss the procedural requirements associated with the Emergency Diesel Generator, including an explanation of major precaution and limitations in the Emergency Diesel Generator procedures
Question Source:	Bank – Braidwood 2020 NRC SRO89
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	43.1/ 45.5 / 45.12 / 45.13
Level:	SRO
Category/System:	06-Emergency Diesel Generator / Conduct of Operations
K/A #:	064 G2.1.9 Ability to direct licensed personnel activities inside the control room (SRO Only).
Importance:	4.5
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the question is asking the SRO what action needs to be directed to the licensed operators in the control room with regards to completing EDG surveillance runs prior to the arrival of a severe storm.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



90 **07-022** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- A Containment Pressure Relief is in progress

Then:

- 2R12A, Containment Atmosphere Radiation Monitor, fails due to a power supply failure
- The Containment Pressure Relief is stopped

Based on the above condition, which of the following is **ONLY** required per the ODCM to recommence the containment pressure relief (assume all other Radiation Monitors are OPERABLE)?

[REFERENCE PROVIDED]

Per the ODCM requirements, the Containment Pressure Relief…

A. may recommence without any compensatory actions.

B. may recommence provided two independent samples of containment are analyzed and two release rate calculations are independently verified.

C. may recommence provided grab samples are taken at least once per 8 hours and analyzed for principal gaseous emitters.

D. may NOT recommence until 2R12A radiation monitor is restored to OPERABLE status.

Answer: A

Explanation / Justification:

A. Correct. Per Table 3.3-13 (Item 3.a) in 3.3.9 of Salem's ODCM, either 2R12A OR 2R41A & D channels is required. Since 2R41 channels are operable (assumption in stem) then ODCM 3.3.9 is MET and no additional action is required to recommence the containment pressure relief.

B. Incorrect. Plausible because this would be the required action (Action 37) if both 2R12A and 2R41A & D channels are INOPERABLE.

C. Incorrect. Plausible because this would be the required action (Action 33) if the SRO incorrectly selects Item 4.a as the affected ODCM instrument. Incorrect in that this is for the Plant Vent Radiation Monitor (R41).

D. Incorrect. Plausible because this is a required action (Action 34) for a containment purge. Incorrect in that this is for a containment pressure relief.

Technical References:	Salem ODCM 3.3.9 (R29)
Proposed References to be provided:	**ODCM 3.3.9 and Bases**
Learning Objective:	NOS05TECHSPEC-12, Objectives 14. Describe the general component and parameter categories that are addressed by Technical Specification Sections 3/4.1 through 3/4.12. **(Licensed Operator & STA only)**
Question Source:	Modified Bank – DC Cook 2018 NRC SRO90
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 43.5 / 43.3 / 45.13 / 9
Level:	SRO
Category/System:	07-Prcess Radiation Monitoring System
K/A #:	073 A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the PRM System, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: PRM component failures
Importance:	3.1
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the question is asking the SRO what action is required per the ODCM when the Containment Radiation Monitor fails. The SRO must be able to evaluate the ODCM actions and determine if the pressure relief may continue and any compensatory actions.

Parent Question: DC Cook 2018 NRC SRO90

Modified Question based on Salem's ODCM actions being different for the failure, therefore the correct answer is different.

90 **ID: 2008NRC-0605-SRO** **Points: 1.00**

SRO Only

Given the following conditions on Unit 1:

- The unit is operating at 100% power.
- A Containment Pressure Relief is in progress.
- VRS-1201 Upper Containment Normal Range Monitor failed due to a power supply failure.
- The Containment Pressure Relief is stopped.

Which ONE of the following describes the restrictions placed on Containment Purge/Pressure Relief Operations?

A. Containment Purge/Pressure Relief operations may NOT be performed until VRS-1201 is restored to operable status.

B. Containment Purge/Pressure Relief operations may continue under administrative controls provided that VRS-1201 is restored to operable status prior to entering Mode 4 following the next refueling outage.

C. Containment Purge/Pressure Relief operations may continue under administrative controls for up to 7 days, provided that area surveys of upper containment are performed at least once every 24 hours.

D. Containment Purge/Pressure Relief operations may continue for up to 48 hours before VRS-1201 is required to be restored to operable status.

Answer: B

Answer Explanation:

A. INCORRECT - Condition A of TS 3.3.6 allows Purge Operations if 2/3 channels per train are operable.

B. CORRECT - Condition A of TS 3.3.6 allows Purge Operations if 2/3 channels per train are operable. The failed channel must be fixed during the next refueling outage.

C. INCORRECT - Sampling every 24 hours is required per TRM 8.3.8 if BOTH VRS-1101 & 1201 are inoperable. Other Channels within TRM 8.3.8 have a 7 day limit.

D. INCORRECT - Purge operations are not limited. These requirements are from the Technical Specification 3.3.6 Condition C.

Additional References: Salem ODCM

PART I RADIOLOGICAL EFFLUENT CONTROLS

Salem REC ODCM Rev. 29
03/21

3/4.3 INSTRUMENTATION

BASES

3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of the "old" 10 CFR Part 20 (ODCM Appendix F). The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

CROSS REFERENCE - TABLES 3.3-13 and 4.3-13

Unit 1 T/S Table Item No.	Instrument Description	Acceptable RMS Channels
1a	Waste Gas Holdup System Noble Gas Activity	1R41A and D(1)(2)
2a	Containment Purge and Pressure - Vacuum Relief Noble Gas Activity	1R12A or 1R41A and D(1)(2)
3a	Plant Vent Header System Noble Gas Activity	1R41A and D(1)(2)
3b	Plant Vent Header System Iodine Sampler (3)	1RME 4, 5 (1R41) or 1RME50, 51 (1R45)
3c	Plant Vent Header System Particulate Sampler (3)	1RME 4, 5 (1R41) or 1 RME50, 51 (1R45)

(1) The channels listed are required to be operable to meet a single operable channel for the ODCM's Minimum Channels Operable" requirement.

(2) 1R41D is the setpoint channel. 1R41A is the measurement channel.

(3) Laboratory analysis of the sampler filters ensures that the limits of ODCM CONTROL 3.11.2.1 are not exceeded. Alarm/trip setpoints do not apply to these passive components.

Additional References: Salem ODCM

3/4.3 INSTRUMENTATION

3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

TABLE 3.3-13: Radioactive Gaseous Effluent Monitoring Instrumentation

Table Notation

ACTION 31 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valving lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 32 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 33 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 8 hours and these samples are analyzed for gaseous principal gamma emitters at the lower limits of detection required in ODCM CONTROL TABLE 4.11-2.A, B, or C within 24 hours. Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 34 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that within 4 hours samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, Containment Pressure Reliefs may be performed provided that prior to initiating the release:

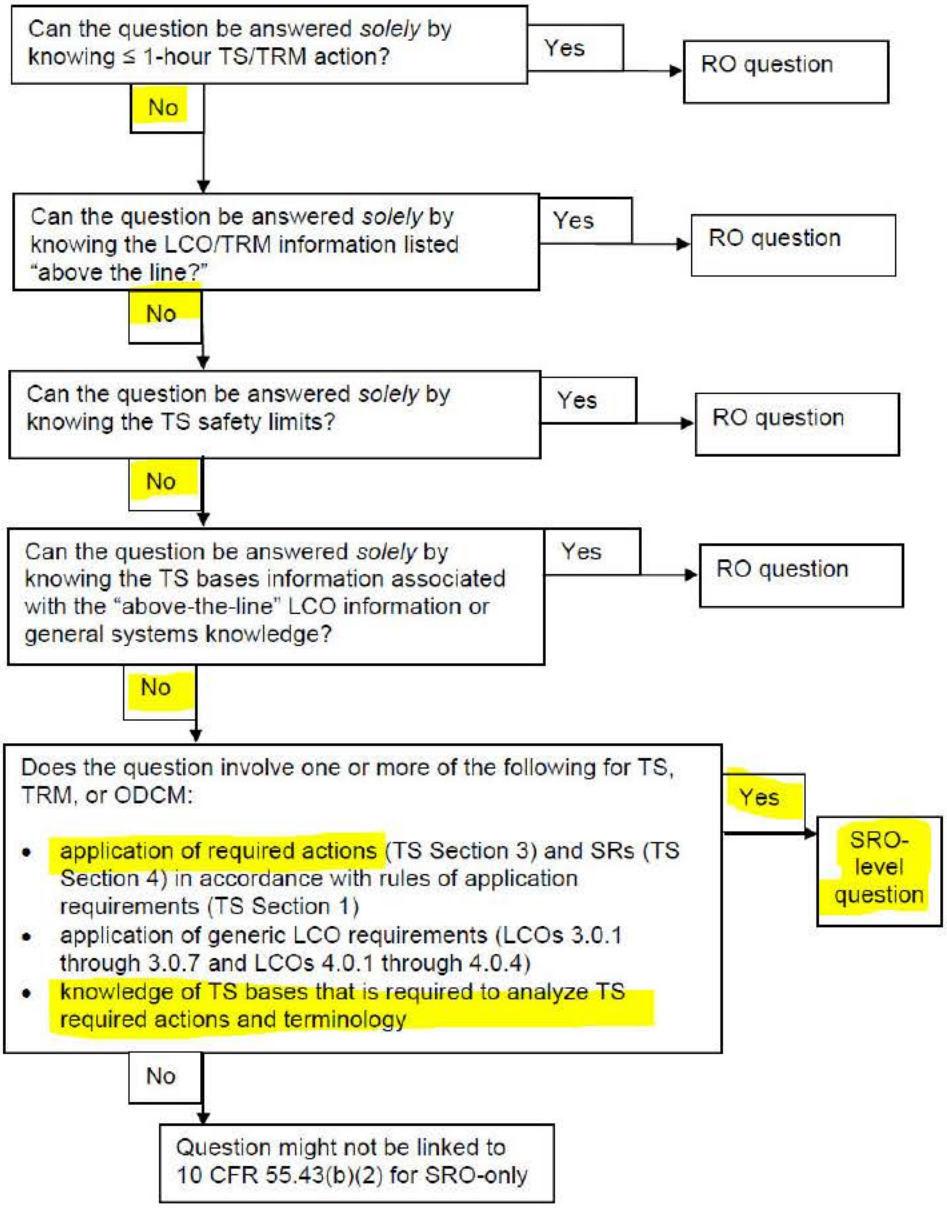
a. At least two independent samples of containment are analyzed, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations.

Otherwise, suspend release of radioactive effluents via this pathway.

SRO Match Justification:

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



91 **02-030** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- 21 Charging Pump is in service
- Surveillance test S2.OP-ST.RC-0007, Seal Injection Flow, was just completed.

The following test data was recorded:

- Charging Header pressure = 2450 psig
- 2CV55 fully open
- 21 RCP seal injection flow = 10 gpm
- 22 RCP seal injection flow = 10.5 gpm
- 23 RCP seal injection flow = 9.8 gpm
- 24 RCP seal injection flow = 10.2 gpm

Based on the surveillance test data, what action is required?

[REFERENCE PROVIDED]

A. Restore the inoperable Charging Pump to Operable status within 72 hours per TS 3.5.2 ECCS Subsystems Tavg > 350 °F.

B. No action is required, the surveillance test is satisfactory.

C. Adjust manual seal injection throttle valves to within the limit within 4 hours as required by TS 3.5.4, Seal Injection Flow.

D. No TS action is required unless the seal injection flows can NOT be adjusted to within the limits within 4 hours.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because the SRO may incorrectly determine that the unsat seal injection flows is a result of a degraded Charging Pump and therefore declare 21 Charging Pump Inoperable and enter TS 3.5.2 Action a.

B. Incorrect. Plausible if the SRO misinterprets the surveillance data and thinks the test data results meet the Tech Spec surveillance requirements for total seal flows.

C. Correct. The surveillance data for total RCP seal injection flows is Unsat due to exceeding the Tech Spec 3.5.4 limit of ≤ 40 gpm. Tech Spec 3.5.4 is not met and the required action is to make adjustments to restore seal injection flows to within limits within 4 hours.

D. Incorrect. Plausible if the SRO misinterprets the note in Tech Spec 3.5.4 about TS 4.0.4 exemption is allowed for up to 4 hours.

Technical References:	TS 3.5.4 and bases
Proposed References to be provided:	**TS 3.5.2 and 3.5.4**
Learning Objective:	NOS05TECHSPEC-12, Objective 14. Describe the general component and parameter categories that are addressed by Technical Specification Sections 3/4.1 through 3/4.12
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	43.2 / 43.5 / 45.12
Level:	SRO
Category/System:	02-Reactor Coolant System
K/A #:	002 G2.2.37 Ability to determine operability or availability of safety related equipment (SRO Only)
Importance:	4.6
Tier/Group	Plant Systems-Tier 2/Group 2

K/A Match: The K/A is matched because the SRO is being asked knowledge on applying TS 3.5.4 with an unsat RCP seal injection surveillance test.

Additional References: Salem TS 3.5.4

EMERGENCY CORE COOLING SYSTEMS

SEAL INJECTION FLOW

LIMITING CONDITION FOR OPERATION

3.5.4 Reactor coolant pump seal injection flow shall be ≤40 gpm with centrifugal charging pump discharge header pressure ≥ 2430 psig and the charging flow control valve full open.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

With seal injection flow not within the limit, adjust manual seal injection throttle valves to give a flow within the limit with the charging pump discharge pressure ≥ 2430 psig and the charging flow control valve full open within 4 hours, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

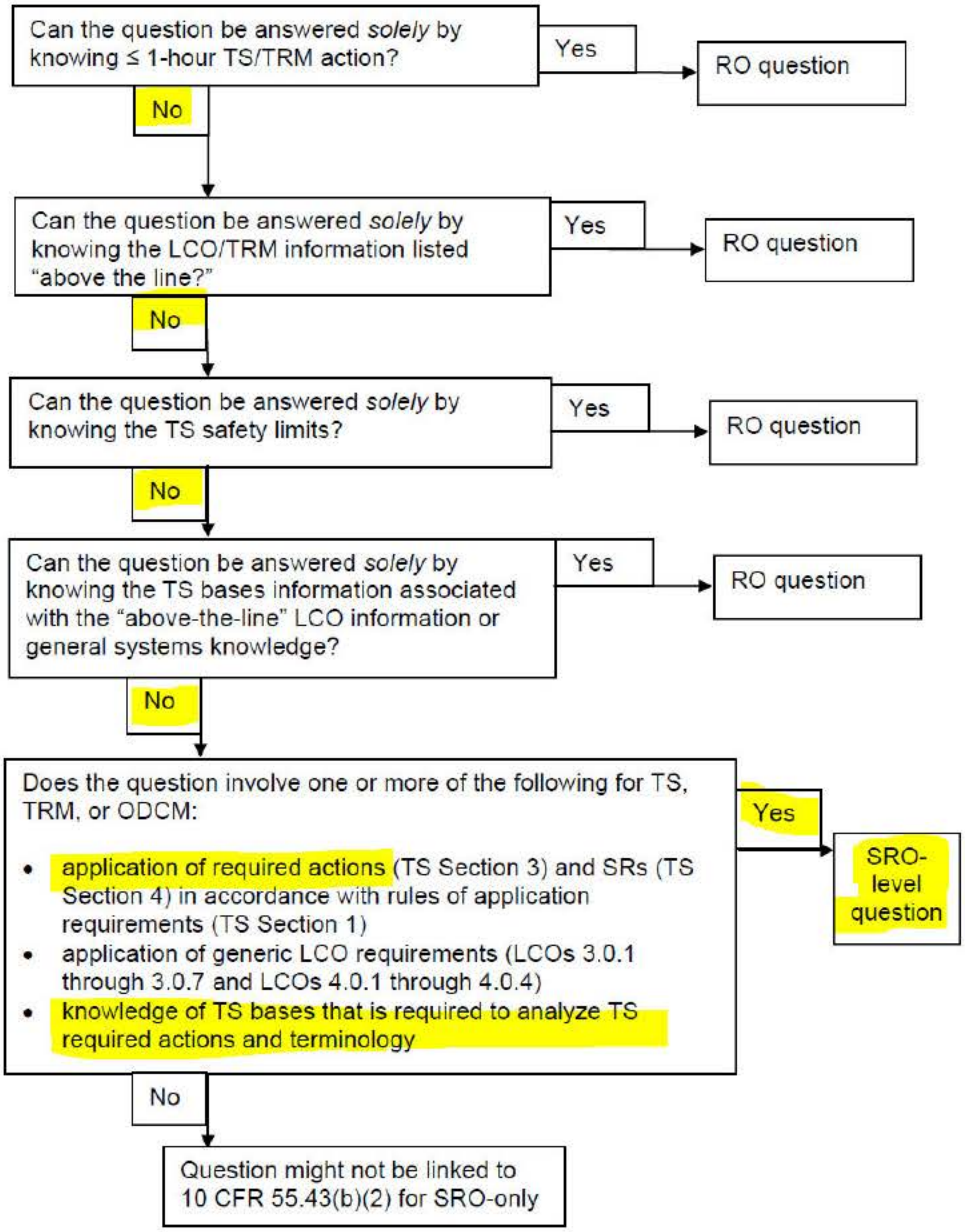
SURVEILLANCE REQUIREMENTS

4.5.4 In accordance with the Surveillance Frequency Control Program, verify manual seal injection throttle valves are adjusted to give a flow within the limit with centrifugal charging pump discharge header pressure ≥ 2430 psig, and the charging flow control valve full open.

The provisions of Specification 4.0.4 are not applicable for entry into Mode 3. This exemption is allowed for up to 4 hours after the Reactor Coolant System pressure stabilizes at 2235 ± 20 psig.

SRO Match Justification:

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



92 **07-023** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- PZR Pressure Channel I failed and has been removed from service per S2.OP-SO.RPS-0003, Placing Pressurizer Channel in Tripped Condition

Then:

- PZR Pressure Channel III fails LOW

Based on the conditions above, which of the following correctly identifies the MOST limiting time requirement for reportability to the NRC?

[REFERENCE PROVIDED]

A. 4 hour report under RAL# 11.3.1

B. 4 hour report under RAL# 11.3.2

C. 8 hour report under RAL# 11.3.3

D. 60 day report under RAL# 11.3.4

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the SRO may believe that this was a valid SI actuation and reportable under RAL 11.3.1 as 4 hour report.

B. Correct. Following the second PZR pressure failing low, a Safety Injection and Rx Trip will occur on low PZR pressure. Since the RCS pressure was not actually low to meet SI actuation, the SI is NOT valid and NOT reportable. However, actuation of RPS is still reportable under RAL 11.3.2 as a 4 hour report based on actuation of the RPS system only, but SI is NOT. Although RPS is a system listed in Technical Bases 11.3.3, it is not reported under this RAL since it will be reported under RAL 11.3.2.

C. Incorrect. Plausible because the SRO may incorrectly interpret that with the SI actuating ECCS, that this is reportable under RAL 11.3.3 as 8 hour report.

D. Incorrect. Plausible because the SRO may believe that the RPS and SI actuations are both INVALID, and since it's a system listed in Technical Bases 11.3.3, then only a 60 day report under RAL 11.3.4 is required.

Technical References:	Salem RAL 11.3 (R4)
Proposed References to be provided:	**Salem RAL 11.3 and Bases**
Learning Objective:	NOS05SENSOR-03.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 /45.6
Level:	SRO
Category/System:	07-Nonnuclear Instrumentation System
K/A #:	016 A2.01 Ability to (a) predict the impacts of the following on the Nonnuclear Instrumentation System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector/transmitter failure
Importance:	3.4
Tier/Group	Plant Systems-Tier 2/Group 2

K/A Match: The K/A is matched because the SRO is being asked to predict the impact on two Nonnuclear Instrumentation (i.e., PZR pressure channel is not related to the detection, control, or indication of neutron flux or power) failures (Rx trip and inadvertent SI) and using the procedure for Reportability Action Levels (RALs) what type of reportability should the SRO make to the NRC.

Additional Reference:

11.0 Reportable Action Levels
11.3 System Actuation

SGS ECG
Rev. 04
Page 1 of 2

REPORTING ACTION LEVELS

Initiating Condition	EVENT THAT RESULTS OR SHOULD HAVE RESULTED IN ECCS DISCHARGE INTO RCS AS RESULT OF A VALID SIGNAL EXCEPT PREPLANNED [10 CFR 50.72(b)(2)(iv)(A)]	ACTUATION OF REACTOR PROTECTION SYSTEM WHEN CRITICAL EXCEPT PREPLANNED [10 CFR 50.72(b)(2)(iv)(B)]	VALID ACTUATION OF LISTED SYSTEM EXCEPT PREPLANNED [10 CFR 50.72(b)(3)(iv)(A)]
MODE	All	1,2	All
RAL #	11.3.1	11.3.2	11.3.3
	IF	IF	IF
	Valid SI Actuation signal received (or demanded).	Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.	Any event or condition that results in valid actuation of any system listed in Technical Basis 11.3.3 except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.
	AND	THEN	THEN
	ANY ECCS pump start or Accumulator depressurization that results in, or should have resulted in, discharge into the reactor coolant system		
	AND		
	Actuation is NOT part of a pre-planned sequence during testing or reactor operation.		
	THEN		
Action Required	Refer to Attachment 14 4-Hour Report		Refer to Attachment 26 8-Hour Report

Additional Reference:

11.0 Reportable Action Levels

11.3 System Actuations

REPORTABLE ACTION LEVEL - 11.3.1

IC ANY EVENT THAT RESULTS OR SHOULD HAVE RESULTED IN ECCS DISCHARGE INTO THE RCS AS THE RESULT OF A VALID SIGNAL EXCEPT WHEN THE ACTUATION RESULTS FROM AND IS PART OF A PRE-PLANNED SEQUENCE DURING TESTING OR REACTOR OPERATION
[10 CFR 50.72(b)(2)(iv)(A)]

RAL

Valid SI Actuation signal received (or demanded)

AND

ANY ECCS Pump start or Accumulator depressurization that results in or should have resulted in, discharge to the RCS

AND

Actuation is NOT part of a pre-planned sequence during testing or reactor operation.

MODE - All

BASIS

A 4-hour report is required for those events that result in either a valid automatic or manual SI actuation or would have resulted in SI actuation if some component had not failed or an operator action had not been taken are reportable. Review RAL 11.3.3 for the list of systems if a valid automatic or manual actuation occurs.

For example, while performing a RCS cooldown following a controlled Reactor Shutdown, a Main Steam Line ΔP SI is inadvertently generated. However, the Charging Pumps fail to start and RCS pressure remains above the SI Pump shutoff head pressure. Although no ECCS discharge to the vessel occurred, the event is reportable.

A system actuation should be apparent at the time of occurrence. Therefore, if all events are reported properly, it is expected that all reports under 10 CFR 50.72 are as a result of an ongoing condition.

A **valid** signal refers to actual plant conditions or parameters satisfying the requirements for SI initiation. Valid actuations also include intentional manual actuations unless the actuation is part of a preplanned test. Excluded from this reporting requirement would be those instances in which instrument drift, spurious signals, human error or other invalid signals caused SI actuation (e.g. jarring a cabinet, an error in the use of jumpers or lifted leads, error in actuation of controls switches, or equipment failures or radio frequency interference).

Additional Reference:

Preplanned actuations are those which are expected to actually occur due to preplanned activities covered by procedures. Such actuations are those for which a procedural step or other appropriate documentation indicates the specific actuation is actually expected to occur. Control room personnel are aware of the specific signal generation before its occurrence or indication in the control room. Manual actuations as directed by abnormal or emergency operating procedures (i.e., not part of a preplanned test or operational evolution) are reportable.

IF the SI Actuation discharges or should have discharged into the RCS as result of an INVALID signal, THEN a report under this RAL is not required.

REFERENCES

SGS UFSAR
10 CFR 50.72(b)(2)(iv)(A)
10 CFR 50.73
NUREG 1022, Rev. 3, section 3.2.6

Additional Reference:

11.0 Reportable Action Levels

11.3 System Actuations

REPORTABLE ACTION LEVEL - 11.3.3

IC VALID ACTUATION OF LISTED SYSTEM EXCEPT PREPLANNED
[10 CFR 50.72(b)(3)(iv)(A)]

RAL

Any event or condition that results in valid actuation of any system listed in Technical Basis 11.3.3 except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation

MODE - All

BASIS

An 8-hour report is required for a valid actuation of any of the systems named in 10 CFR 50.72(b)(3)(iv)(B) unless the actuation resulted from and was part of a pre-planned sequence during testing or reactor operation. Except for critical scrams (RAL 11.3.2), invalid actuations are not reportable by telephone under 10 CFR 50.72. If an 8-hour ENS report is needed due to a valid system actuation concurrent with an ECCS or RPS actuation, it can be included with the 4-hour ENS report made under 10 CFR 50.72(b)(2)(iv)(A) or (B).

The system actuation flow chart provides guidance to determine reportability.

REFERENCES

10 CFR 50.72(b)(3)(iv)
10 CFR 50.73(a)(2)(iv)
NUREG-1022, Rev. 3, section 3.2.6

Additional Reference:

SGS EAL/RAL Technical Basis

Applicable Systems and Components

NOTE: Numbers in parentheses indicate UFSAR Chapter

Reactor Protection System (unless reported under RAL 11.3.2)

Containment Systems (6.2)
- Containment Heat Removal (6.2.2)
- Containment Isolation System* (6.2.4)

ECCS (6.3)
- Residual Heat Removal
- Safety Injection System

Plant Systems
- Auxiliary Feedwater

Emergency AC Electrical Power (8.3)

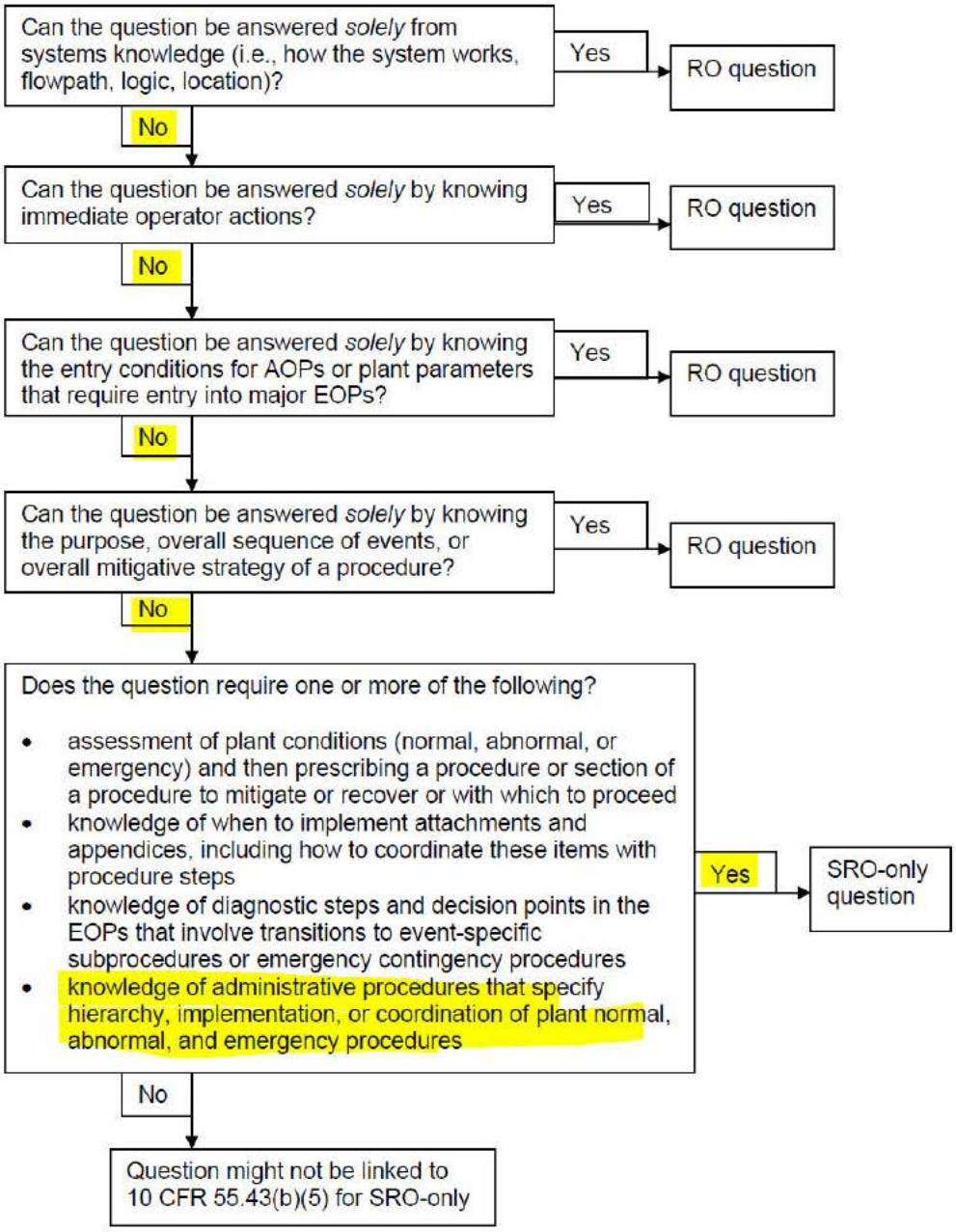
* Containment isolation valves in more than one system or multiple MSIVs

REFERENCES

SGS UFSAR
10 CFR 50.72(b)(3)(iv)(A)
10 CFR 50.73
NUREG 1022, Rev. 3, section 3.2.6

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



93 **08-020** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- 21A CW Pump is C/T for waterbox cleaning

Then:

- The 4KV Infeed breaker to the 24 CW bus fails opens

What action will the CRS direct the crew?

A. Power operation at 100% may continue since the 24 CW Bus will auto transfer to the 23 CW Bus.

B. Initiate a load reduction to less than 49% power and trip the main turbine per S2.OP-AB.TRB-0001, Turbine Trip Below P-9.

C. Initiate a load reduction to less than or equal to 83% power per S2.OP-AB.CW-0001, Circulating Water Malfunction.

D. Trip the reactor and enter 2-EOP-TRIP-1, Reactor Trip or Safety Injection, per S2.OP-AB.CW-0001, Circulating Water Malfunction.

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the SRO may believe that the CW Bus Auto Transfer scheme will transfer the 24 CW bus to 23 CW bus without losing power to the 'B' Circulators and therefore, no impact to the current power level.

B. Incorrect. Plausible because the SRO may believe that removing the main turbine from service is allowed using S2.OP-AB.TRB-0001.

C. Incorrect. Plausible because a loss of both 21A and 22A circulators requires a load reduction to less than or equal to 83% power.

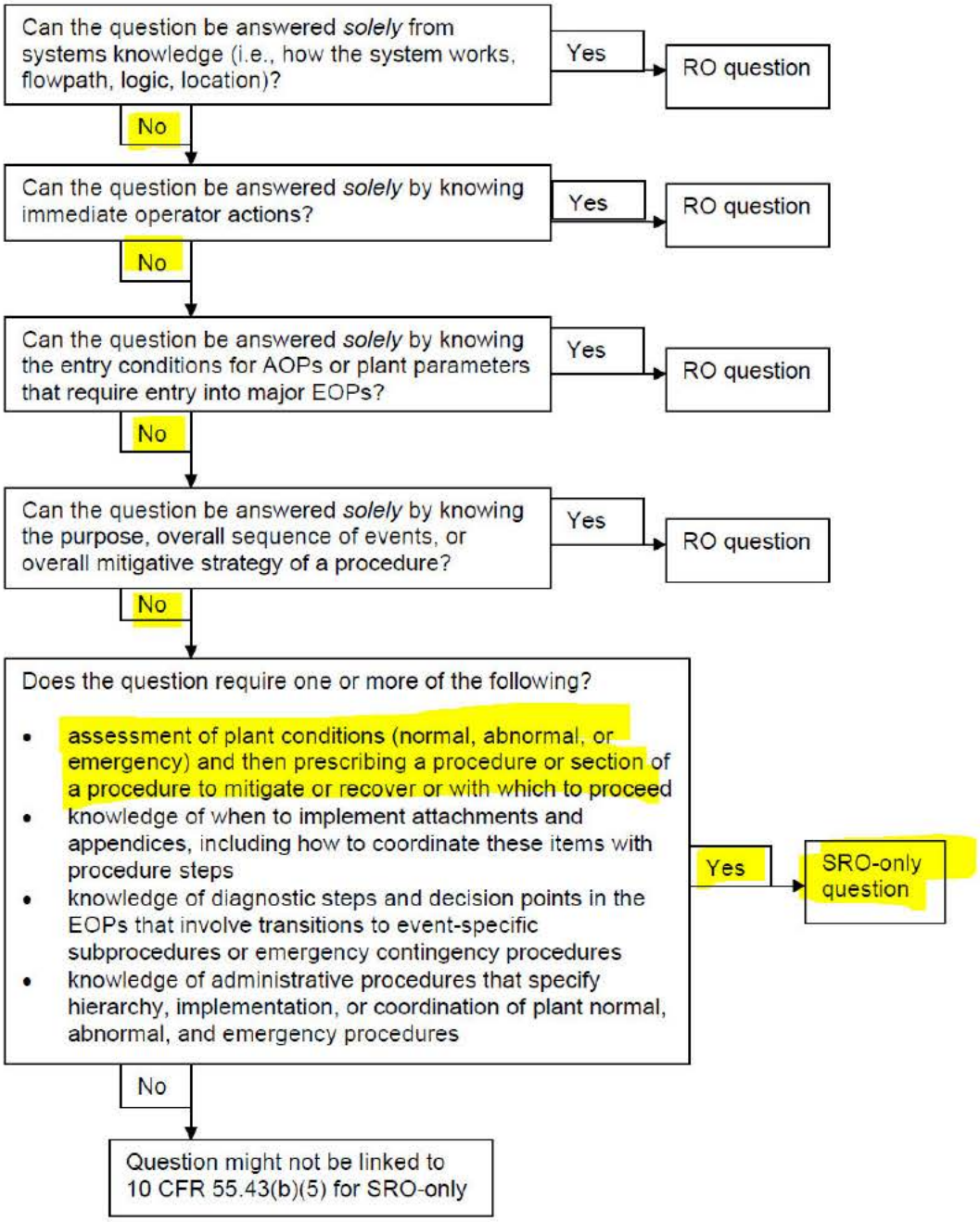
D. Correct. When the infeed breaker (24CW9AD) fails open, the 24 CW bus will de-energize, when the bus voltage is < 70% the bus will transfer automatically to the 23 CW bus. However, all the 'B' Circulator (21B, 22B, and 23B) breakers will trip open on undervoltage and must be manually started following the bus transfer. The traveling screen power will be supplied from the 23 CW bus providing power to the 'A' Circulators. Per S2.OP-AB.CW-0001 Attachment 1, with 4 or more circulators out of service with power greater than or equal to P-10 (10%), then trip the reactor and enter 2-EOP-TRIP-1.

Technical References:	S2.OP-AB.CW-0001 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05ABCW01-13, Objective 4. Describe, in general terms, the actions taken in S2.OP-AB.CW-0001 and the bases for the actions in accordance with the Technical Bases Document.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 43.5 / 45.3 / 45.13
Level:	SRO
Category/System:	08-Circulating Water System
K/A #:	075 A2.02 Ability to (a) predict the impacts of the following on the CW System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of circulating water pumps
Importance:	3.4
Tier/Group	Plant Systems-Tier 2/Group 2

K/A Match: The K/A is matched because the SRO is being asked to assess plant conditions and then select the specific procedural to mitigate the loss of 4 or more Circulators.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



94 **14-082** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- The crew is at minimum shift complement

Then:

- A shift change is in progress
- An on-duty Unit 2 Reactor Operator reports that his relief is unable to report to work
- There are NO available qualified licensed ROs on-site

In accordance with OP-AA-101-111, Roles and Responsibilities of On-Shift Personnel, which of the following completes both statements?

1. The on-duty Unit 2 RO shift position __(1)__ be unmanned upon shift change.
2. The shift complement may be one less than the minimum requirements for a period not to exceed __(2)__.

A. (1) can (2) 2 hours

B. (1) can (2) 1 hour

C. (1) can NOT (2) 1 hour

D. (1) can NOT (2) 2 hours

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the SRO may misinterpret the requirements in OP-AA-101-111 and believe that since 2 hours is allowed to fill the unexpected absence that the on duty RO can leave at the end of the shift.

B. Incorrect. Plausible because the SRO may misinterpret the requirements in OP-AA-101-111 and believe that since 2 hours is allowed to fill the unexpected absence that the on duty RO can leave at the end of the shift.

C. Incorrect. Plausible because 1 hour requirements are used in Tech Specs and the SRO may believe that this 1 hour action is warranted.

D. Correct. Per OP-AA-101-111, Attachment 2 Note 1, the Salem Shift complement may be one less than the minimum requirements for a period not to exceed 2 hours to accommodate the unexpected absence of on duty shift members provided that immediate action is taken to restore the Shift Complement to within the minimum requirements. Also, this provision does not permit any shift position to be unmanned upon shift change due to an oncoming shift member being late or absent. The preferred action is to hold the off-going shift personnel until a relief is available.

Technical References:	OP-AA-101-111 (R11), TS 6.2.2
Proposed References to be provided:	None
Learning Objective:	NOS05CONDOP-14, Objective 5.g. Describe requirements for the following Control Room or Field Activities in accordance with applicable Conduct of Operations Manual Administrative: Procedures: Shift Relied and Turnovers
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 43.5 / 45.12
Level:	SRO
Category/System:	14-Conduct of Operations
K/A #:	G2.1.5 Ability to use procedures related to shift staffing, such as minimum crew complement or overtime limitations (reference potential).
Importance:	3.9
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the candidate requires knowledge of administrative procedures related to minimum shift staffing requirements. The question is asking the SRO the required time to fill a vacant licensed position per TS and whether the on duty shift personnel can leave if his relief is absent.

Additional Reference:

Attachment 2, Salem Shift Complement

[CD 419Y, 70145753, T/S 6.2.2] (Notes 1, 6, and 7)

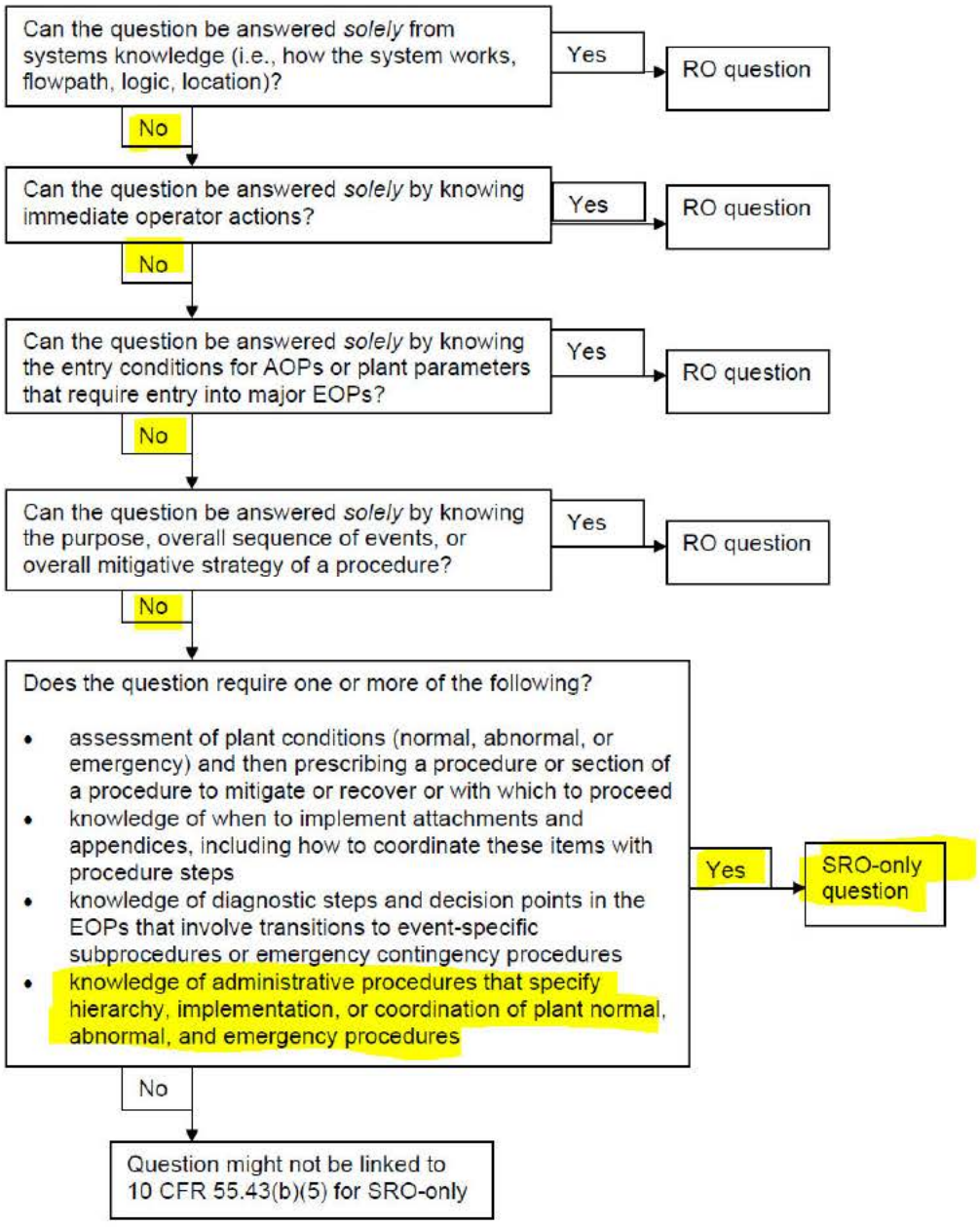
	Salem Unit 1 Mode	**1-4**	**1-4**	**5-6, D**	**5-6, D**
	Salem Unit 2 Mode	**1-4**	**5-6, D**	**1-4**	**5-6, D**
	SM	1	1	1	1
	CRS	2	2	2	1
	STA	1	1	1	1
	OSCC (Note 4)	1	1	1	1
	RO/PO	4	4	4	4
	NEO	6	6	6	6
	Communicator (Note 2)	2	2	2	2
CD-252X	Shift Electrician	1 (each)	1 (each)	1 (each)	1 (each)
CD-252X	Shift I&C Technician	1 (each)	1 (each)	1 (each)	1 (each)
CD-252X	RP Technician (Note 3)	2	2	2	2
	Fire Brigade Liaison (FBL) (Note 8)	1	1	1	1
	Fire Brigade	IAW FP-AA-012			
	Debris Removal Operator (DRO) (Note 5)	1	1	1	1
	Towing Operator (TO) (Note 5)	1	1	1	1

This attachment combines the minimum shift staffing requirements of the Technical Specifications, Emergency Planning, Appendix R Simultaneous Alternate Shutdown, and Beyond Design Basis External Event (BDBEE) FLEX Strategies. Clarifying statements are identified separately below:

1. Except for the SM, the Salem Shift Complement may be one less than the minimum requirements for a period not to exceed 2 hours to accommodate the unexpected absence of on duty shift members provided that immediate action is taken to restore the Shift Complement to within the minimum requirements of the tables. This provision does not permit any shift position to be unmanned upon shift change due to an oncoming shift member being late or absent. The preferred action is to hold over the off-going shift personnel until a relief is available. **[CD 252X]**

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



95 **14-083** **Points: 1.00**

Given:

- Salem Unit 2 is in MODE 6
- Conditions are established per S2.OP-IO.ZZ-0010, Spent Fuel Movements, to support fuel handling operations in the Fuel Handling Building (FHB)
- Movement of irradiated fuel in the FHB is in progress

Then:

- FHB console alarm, BLDG AIR D/P LO, is Alarming
- No operator action has been taken

Based on the above condition and per S2.OP-AR.ZZ-0011, 2CC1 Console Alarm, which of the following completes both statements?

1) What is the NEXT required action?
2) If the BLDG AIR D/P LO alarm is INVALID and will NOT clear, then the local indicator __(2)__ be used to continuously monitor FHB D/P.

A. (1) Validate the alarm using the local indicator.
(2) may NOT

B. (1) Suspend movement of irradiated fuel assemblies.
(2) may

C. (1) Suspend movement of irradiated fuel assemblies.
(2) may NOT

D. (1) Validate the alarm using the local indicator.
(2) may

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because validating the console alarm is an action performed. Incorrect in that the first action is to suspend fuel moves in the FHB and then validate the alarm.

B. Correct. Per S2.OP-AR.ZZ-0011 for BLDG AIR D/P LO, if movement of irradiated fuel assemblies are in progress, then suspend movement of irradiated fuel assemblies in the FHB. The next action is to validate the alarm by checking the local indicator. If the alarm is considered invalid, then compensatory actions are required. Per S2.OP-IO.ZZ-0010, if the BLDG AIR D/P LO is not clear, then the local indicator may be used to continuously monitor FHB D/P.

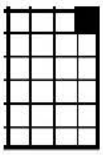
C. Incorrect. Plausible because the SRO may believe that without the console alarm operable, then fuel moves inside the FHB cannot continue until the alarm function is restored. Incorrect in that procedures allow the use of the local indicator so long as it's continuously monitored.

D. Incorrect. Plausible because validating the console alarm is an action performed. Incorrect in that the first action is to suspend fuel moves in the FHB and then validate the alarm.

Technical References:	S2.OP-IO.ZZ-0010 (R38), S2.OP-AR.ZZ-0011 (R64)
Proposed References to be provided:	None
Learning Objective:	NOS05IOP010-05, Objective 7. Evaluate the actions required by TSs or TRM during performance of S1/S2.OP-IO.ZZ-0010, Spent Fuel Pool Manipulations if any of the following limits are exceeded: FHB DP greater than -0.125 (less negative).
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	43.7
Level:	SRO
Category/System:	14-Conduct of Operations
K/A #:	G2.1.35 Knowledge of the fuel handling responsibilities of SROs (SRO Only).
Importance:	3.9
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the SRO is being asked knowledge of fuel handling responsibilities associated with the movement of irradiated fuel in the FHB and what action is required when the FHB building D/P alarm is in and not valid.

Additional Reference: S2.OP-AR.ZZ-0011 (2CC1)

SYSTEM FUEL HANDLING BUILDING VENTILATION Exhaust Filter Trains	BLDG AIR D/P LO
	
BEZEL Controls	

DEVICES: 2PD-6531

SETPOINT: -0.160" H_2O

1.0 **CAUSE(S)**:

1.1 Loss of slight negative pressure in the Fuel Handling Building

2.0 **AUTOMATIC ACTIONS**:

2.1 None

3.0 **OPERATOR ACTIONS**:

3.1 IF movement of irradiated fuel assemblies is in progress
THEN **PERFORM** the following:

A. **SUSPEND** movement of irradiated fuel assemblies in the FHB.

B. **SEND** an Operator to Panel 436-2 (130' FHB):

1. **CHECK** alarm setpoint at 2PD6531, FHB VENT LO DP SW.

2. **VALIDATE** the BLDG AIR D/P LO alarm using local indicator 2PD6531.

C. IF alarm is validated, **DEPRESS** INLET VANE OPEN bezel for 21 AND 22 FHB EXHAUST FANS.

D. IF alarm clears OR FHB can be maintained at a slight negative pressure
THEN movement of irradiated assemblies may continue.

E. IF alarm remains illuminated with FHB at negative pressure
THEN **IMPLEMENT** compensatory actions as required.

Additional Reference:

S2.OP-IO.ZZ-0010(Q)

ATTACHMENT 3 Sheet ___ of ___
(Page 1 of 3)

REVIEWS FOR CONTINUING FUEL HANDLING

1.0 SHIFT REVIEW:

1.1 **REVIEW** entire procedure
AND **ENSURE** ALL conditions still valid.

1.2 **ENSURE** the following:
A. BOTH FHB Exhaust Fans AND one Supply Fan must be AVAILABLE AND operating.
B. Fuel Handling Building is maintained at slight negative pressure with respect to atmosphere pressure.
C. BLDG AIR D/P LO alarm is clear OR local indicator 2PD6531 being used to continuously monitor FHB ΔP.

1.3 **ENSURE** ALL Open Items listed in Section 2.0 of attachment resolved prior to resuming Spent Fuel Pool Manipulations.

SRO Match Justification:

f. Procedures and Limitations Involved in Initial Core Loading, Alterations in Core Configuration, Control Rod Programming, and Determination of Various Internal and External Effects on Core Reactivity [10 CFR 55.43(b)(6)]

Some examples of SRO-only examination items for this topic include the following:

- evaluation of core conditions and emergency classifications based on core conditions
- administrative requirements associated with low-power physics testing processes
- administrative requirements associated with refueling activities, such as approvals required to amend core loading sheets or administrative controls of potential dilution paths or activities
- administrative controls associated with the installation of neutron sources
- knowledge of TS bases for reactivity controls

g. Fuel Handling Facilities and Procedures [10 CFR 55.43(b)(7)]

Some examples of SRO-only examination items for this topic include the following:

- refuel floor SRO responsibilities
- assessment of fuel handling equipment SR acceptance criteria
- prerequisites for vessel disassembly and reassembly
- decay heat assessment
- assessment of SRs for the refueling mode
- reporting requirements
- emergency classifications

96 **14-084** **Points: 1.00**

Per WC-AA-101, On-Line Work Management Process, which of the following choices are responsibilities of the Shift Manager in support of online maintenance?

1. Ensures site FLEX capability is maintained.
2. Ensures risk is reassessed when an emergent condition results in a plant configuration not previously assessed.
3. Ensures Protected Equipment barriers are applied per OP-AA-108-116, Protected Equipment Program.
4. Manages special or complex projects during on-line implementation.

A. 1, 2, 3, and 4

B. 1, 2, and 3 Only

C. 2 and 3 Only

D. 1 and 4 Only

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the SRO may believe that the SM has this responsibility based on the complex project. Incorrect in that per WC-AA-101, the responsibility belongs to the Online Maintenance Project Coordinator.

B. Correct. Per WC-AA-101, step 3.6, the Shift Manager responsibilities include the following: 3.6.1 Ensures shift operations reassesses risk if an emergent condition results in a plant configuration that has not been previously assessed, 3.6.2 Ensures shift operations apply Protected Equipment barriers IAW OP-AA-108-116, 3.6.3 Ensures fire in a(4) Risk Management Actions are implemented as applicable IAW the appropriate site specific procedure, 3.6.4 Ensures the site FLEX capability is maintained.

C. Incorrect. Plausible because these are responsibilities of the SM. Incorrect in that choice 1 is also a responsibility of the SM.

D. Incorrect. Plausible because choice 1 is a responsibility of the SM. Incorrect in that per WC-AA-101, the responsibility belongs to the Online Maintenance Project Coordinator.

Technical References:	WC-AA-101 (R30)
Proposed References to be provided:	None
Learning Objective:	NOS05WORK00-07, Objective4. Describe the conditions under which the SM/CRS permission must be obtained prior to starting work, in accordance with WC-AA-10, Work Management Process Description, and WC-AA-101, On-Line Work Management Process.
Question Source:	Modified - Byron 2017 NRC SRO96
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	SRO
Category/System:	14-Equipment Control
K/A #:	G2.2.19 Knowledge of maintenance work order requirements.
Importance:	3.4
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the question is asking the SRO knowledge on administrative controls for performing maintenance on systems and plant components per WC-AA-101.

Additional Reference: WC-AA-101 page 6

3.4 **Work Week Manager (WWM)**

3.4.1. Ensures that a risk assessment of planned work has been performed and evaluates the results against the criteria of Attachment 3, Configuration Management Criteria

3.4.2. Oversees the preparation and coordination activities of Project Coordinators required by this procedure.

3.4.3. Ensures completion and approval of Attachment 2, On-Line Maintenance Approval Form, when required.

3.5 **Online Maintenance Project Coordinator**

3.5.1. Designated by Cycle Manager or WWM to manage special or complex projects during on-line operation.

3.5.2. Serves as the single point of contact between the Shift, Work Week Manager, and work groups.

3.6 **Shift Manager**

3.6.1. Ensures shift operations reassesses risk if an emergent condition results in a plant configuration that has not been previously assessed.

3.6.2. Ensures shift operations apply Protected Equipment barriers IAW OP-AA- 108-116, "Protected Equipment Program".

3.6.3. Ensures fire in a(4) Risk Management Actions are implemented as applicable IAW the appropriate site specific procedure.

3.6.4. Ensures the site FLEX capability is maintained.

3.7 **Operations Services Manager**

3.7.1. Ensures any threats to generation are provided to the Electrical System Operations Center (ESOC) as prescribed by OP-AA-108-107-1001, Electric System Emergency Operations and Electric Systems Operator Interface.

3.7.2. Ensures planned load reductions are coordinated with the ESOC.

Parent Question: Byron 2017 NRC SRO96

Modified question by adding a third correct choice and some of the choices from the parent question are different that PSEG site procedure.

96 **ID: 2017 NRC Q96C** **Points: 1.00**

Per WC-AA-101, ONLINE WORK CONTROL PROCESS, which of the following are responsibilities of Operations Shift Management?

1. Audits the work planning and scheduling processes to ensure maintenance is completed.
2. Ensures appropriate actions are taken to mitigate online risk.
3. Responsible for declaring Structure, System, and Component (SSC) functionality/availability.
4. Ensures project critiques are performed for projects requiring a fragnet.

A. 1 and 2

B. 2 and 3

C. 3 and 4

D. 4 and 1

Answer: B

SRO Match Justification:

c. *Facility Licensee Procedures Required to Obtain Authority for Design and Operating Changes in the Facility [10 CFR 55.43(b)(3)]*

Some examples of SRO-only examination items for this topic include the following:

ES-4.2, Page 21 of 29

- screening and evaluation processes under 10 CFR 50.59, "Changes, tests, and experiments"
- administrative processes for temporary modifications
- administrative processes for disabling annunciators
- administrative processes for the installation of temporary instrumentation
- processes for changing the plant or plant procedures

97 **14-085** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

At Time 0500:

- The crew enters S2.OP-AB.RC-0001, RCS Leak, for a suspected RCS leak

At Time 0505:

- The estimated leak rate is 12 gpm
- The leak is identified from the 2CV6, Letdown Relief Valve

At Time 0516:

- Letdown is isolated and the leakage has stopped

Which of the following completes both statements?

1. At time 0505, what is the status of TS 3.4.7.2, Operational Leakage?
2. At time 0516, the CRS _____ required to make an Emergency Declaration.

[REFERENCE PROVIDED]

A. (1) NOT Met
(2) is NOT

B. (1) NOT Met
(2) is

C. (1) Met
(2) is

D. (1) Met
(2) is NOT

Answer: D

Explanation / Justification:

A. Incorrect. Part 1 is incorrect. Plausible because the SRO may believe that the CVCS system is part of the RCS system. Incorrect in that the CVCS system is considered an auxiliary system and not part of the RCS system. Part 2 is correct.

B. Incorrect. Part 1 is incorrect. Plausible because the SRO may believe that the CVCS system is part of the RCS system. Incorrect in that the CVCS system is considered an auxiliary system and not part of the RCS system. Part 2 is incorrect. Plausible because the SRO may incorrectly determine that the RCS leakage exceeded the criteria EAL SU5.1.

C. Incorrect. Part 1 is incorrect. Plausible because the SRO may believe that the CVCS system is part of the RCS system. Incorrect in that the CVCS system is considered an auxiliary system and not part of the RCS system. Part 2 is incorrect. Plausible because the SRO may incorrectly determine that the RCS leakage exceeded the criteria EAL SU5.1.

D. Correct. Per the bases of S2.OP-AB.RC-0001, a RCS leak on the CVCS system that can be isolated does not require entry into TS 3.4.7.2 or requires an Emergency Notification (UE EAL SU5.1).

Technical References:	S2.OP-AB.RC-0001 (R18), EP-SA-325-218 (R0)
Proposed References to be provided:	**EP-SA-325-218 (Bases) and EP-SA-325-118 (EAL SU5.1)**
Learning Objective:	NOS05ABRC01-04, Objective 2. Describe, in general terms, the actions taken in S1/S2.OP-AB.RC-0001 and the bases for the actions in accordance with the Technical Bases Document
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 43.5 / 45.12
Level:	SRO
Category/System:	14-Equipment Control
K/A #:	G2.2.44 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.
Importance:	4.4
Tier/Group	Generic Knowledge and Abilities-Tier 3

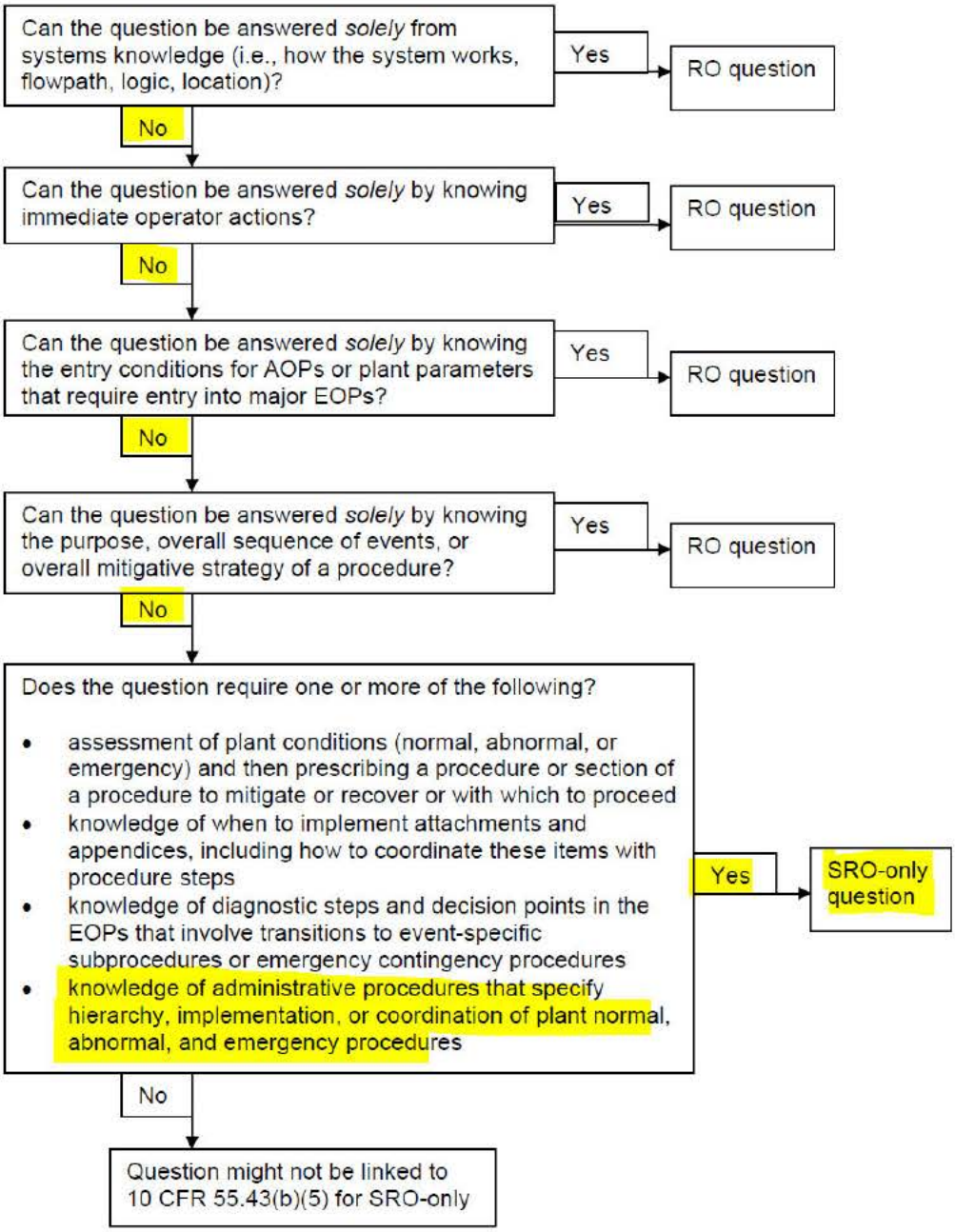
K/A Match: The K/A is matched because the question is asking the SRO to verify the status of the RCS leak on the CVCS system and determine whether or not this event is reportable to the States.

Additional Reference: S2.OP-AB.RC-0001 bases

At Step 3.24, the CRS/SM is notified to refer to the Event Classification Guide and to Technical Specifications. RCS Leakage definition for ECG classification purposes is "Any leakage that is unisolable should be considered Reactor Coolant System (RCS) Leakage when reviewing the Salem ECG for classification assessment". Leakage from Chemical Volume and Control System (CVCS) letdown piping, or components such as the 2CV6 relief valve or vent and drain valves associated with CVCS demineralizer vessels, is NOT considered Reactor Coolant System (RCS) Leakage. The CVCS system piping and components are capable of being isolated from the RCS piping and are classified in the Salem UFSAR as auxiliary to the reactor coolant system pressure boundary. In addition, Industry OE shows several plants have misclassified leakage from the CVCS as RCS Leakage and later retracted the TS Action Statement entry and Event notification. As a result, the entry into the Salem TS Action Statement for Identified Leakage is NOT required due to the source of the leakage being part of an isolable portion of the CVCS (non-RCS Leakage) and NOT from the RCS as defined in Technical Specifications.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



98 **14-086** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- A release of 21 CVCS Monitor Tank needs to be performed
- 2R18, Radioactive Liquid Effluent Monitor, was declared inoperable 3 days ago

Which of the following completes both statements?

1. Before performing the liquid radioactive waste release, the __(1)__authorizes the discharge.
2. Based on the above condition, the ODCM requires that _____(2)_____.

Note: ODCM = Offsite Dose Calculation Manual

A. (1) Shift Manager/Control Room Supervisor
(2) at least two independent samples of the monitor tank are analyzed prior to the release

B. (1) Shift Manager/Control Room Supervisor
(2) continuous surveys of the discharge piping are performed during the release

C. (1) Radiation Protection Manager
(2) continuous surveys of the discharge piping are performed during the release

D. (1) Radiation Protection Manager
(2) at least two independent samples of the monitor tank are analyzed prior to the release

Answer: A

Explanation / Justification:

A. Correct. Per the ODCM 3.3.8 Radioactive Liquid Effluent Monitoring Instrumentation, with 2R18 inoperable action 26 applies which requires at least two independent samples are analyzed and at least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving. Per S2.OP-SO.WL-0001, step 2.5 the SM/CRS authorizes the release.

B. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because surveys of piping are performed but for different reason. Incorrect in that the ODCM requires two independent samples analyzed prior to the release which is verified by Chemistry.

C. Incorrect. Part 1 is incorrect. Plausible because the SRO may confuse the responsibility of a radioactive liquid release with a Rad Pro function. Part 2 is incorrect. Plausible because surveys of piping are performed but for different reason. Incorrect in that the ODCM requires two independent samples analyzed prior to the release which is verified by Chemistry.

D. Incorrect. Part 1 is incorrect. Plausible because the SRO may confuse the responsibility of a radioactive liquid release with a Rad Pro function. Part 2 is correct.

Technical References:	S2.OP-SO.WL-0001 (R32), ODCM 3.3.8 (R29)
Proposed References to be provided:	None
Learning Objective:	NOS05TECHSPEC-12, Objective 14. Describe the general component and parameter categories that are addressed by Technical Specification Sections 3/4.1 through 3/4.12Question Source: Bank - Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.13 / 43.4 / 45.10
Level:	SRO
Category/System:	14-Radiation Control
K/A #:	G2.3.6 Ability to approve release permits.
Importance:	3.8
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the SRO requires knowledge of authorizing releases of liquid radioactive waste and additional requirements to perform the release if the radiation monitor is inoperable.

SRO Match Justification:

d. Radiation Hazards that May Arise during Normal and Abnormal Situations, including Maintenance Activities and Various Contamination Conditions [10 CFR 55.43(b)(4)]

Some examples of SRO-only examination items for this topic include the following:

- process for gaseous/liquid release approvals (i.e., release permits)
- analysis and interpretation of radiation and activity readings as they pertain to the selection of administrative, normal, abnormal, and emergency procedures
- analysis and interpretation of coolant activity, including comparison to emergency plan criteria or regulatory limits (or both)
- process for authorizing emergency exposure limits

99 **15-044** **Points: 1.00**

Given:

- The STA is monitoring Critical Safety Function Status trees

Which of the following completes the statement?

If Containment Radiation Monitors __(1)__ exceeds 2 R/HR, the CRS __(2)__ 2-EOP-FRCE-3, Response to High Containment Radiation Level.

A. (1) R10A/B
(2) must GO TO

B. (1) R44A/B
(2) must GO TO

C. (1) R44A/B
(2) may elect to GO TO

D. (1) R10A/B
(2) may elect to GO TO

Answer: C

Explanation / Justification:

A. Incorrect. Part 1 and 2 are incorrect.

B. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because if the SRO believes a Red or Purple path exists, then entry into the FRP is required. Incorrect in that a yellow path exist and entry is optional.

C. Correct. Per 2-EOP-CFST-1, for containment environment, the R44A and R44B radiation monitors are used to determine whether entry into 2-EOP-FRCE-3 is required if radiation levels are above 2R/HR. FRCE-3 is a yellow path procedure, not RO knowledge, and the SRO must apply the rules of usage for FRPs to determine how the procedure is performed. Yellow path procedures are optional.

D. Incorrect. Part 1 is incorrect. Plausible because the R10A/B radiation monitors are located inside containment. Incorrect in that the R44A/B are the high range radiation monitors used in the CFSTs. Part 2 is correct.

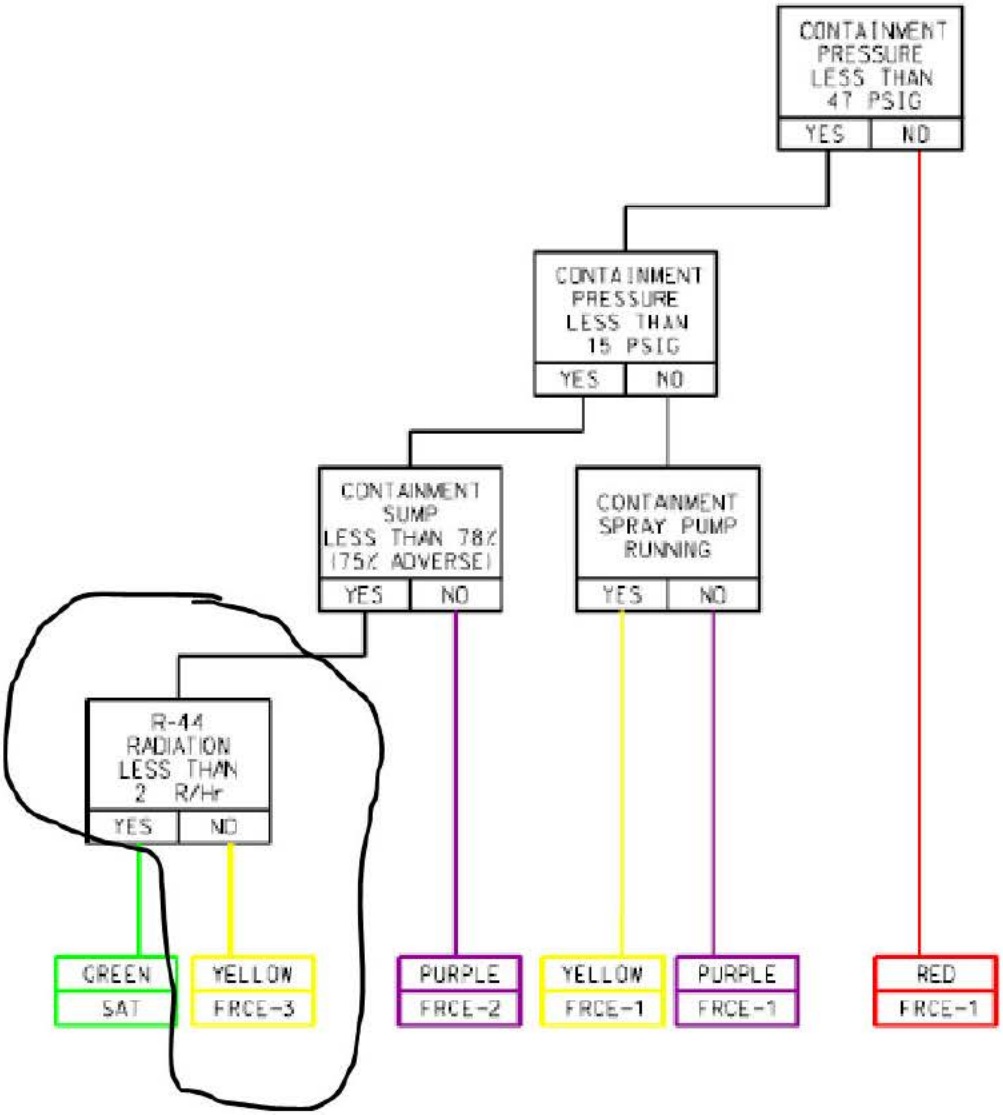
Technical References:	2-EOP-CFST-1 (R41), OP-SA-108-101-2002 (R0)
Proposed References to be provided:	None
Learning Objective:	NOS05TRP001-09, Objective 10, Given EOP-CFST-1 and a set of plant conditions, determine whether Functional Restoration Procedures (FRPs) should be implemented and/or terminated, in accordance with OP-SA-108-101-2002 and EOP-CFST-1.
Question Source:	Bank – Diablo Canyon 2018 NRC SRO85
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.7 / 45.7 / 45.8
Level:	SRO
Category/System:	15-Emergency Procedures/Plan
K/A #:	G2.4.2 Knowledge of system setpoints, interlocks, and automatic actions associated with emergency and abnormal operating procedure entry conditions.
Importance:	4.6
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the question is asking the SRO which component is used to determine entry into a FRP procedure and based on the status of the trees, determine whether entry is required or optional.

Additional References: 2-EOP-CFST-1

FIGURE 5
CONTAINMENT ENVIRONMENT STATUS TREE

EOP-CFST-1



Additional Reference:

4.2.14. If a FRG is being performed, and a higher priority RED or PURPLE path comes in, the current FRG should be suspended and a transition made to the higher priority FRG unless stated otherwise in the applicable guideline. After the new FRG has been completed and guidance is provided to "return to procedure and step in effect," the operator should go back to the previous FRG which had been implemented, and was therefore the procedure in effect.

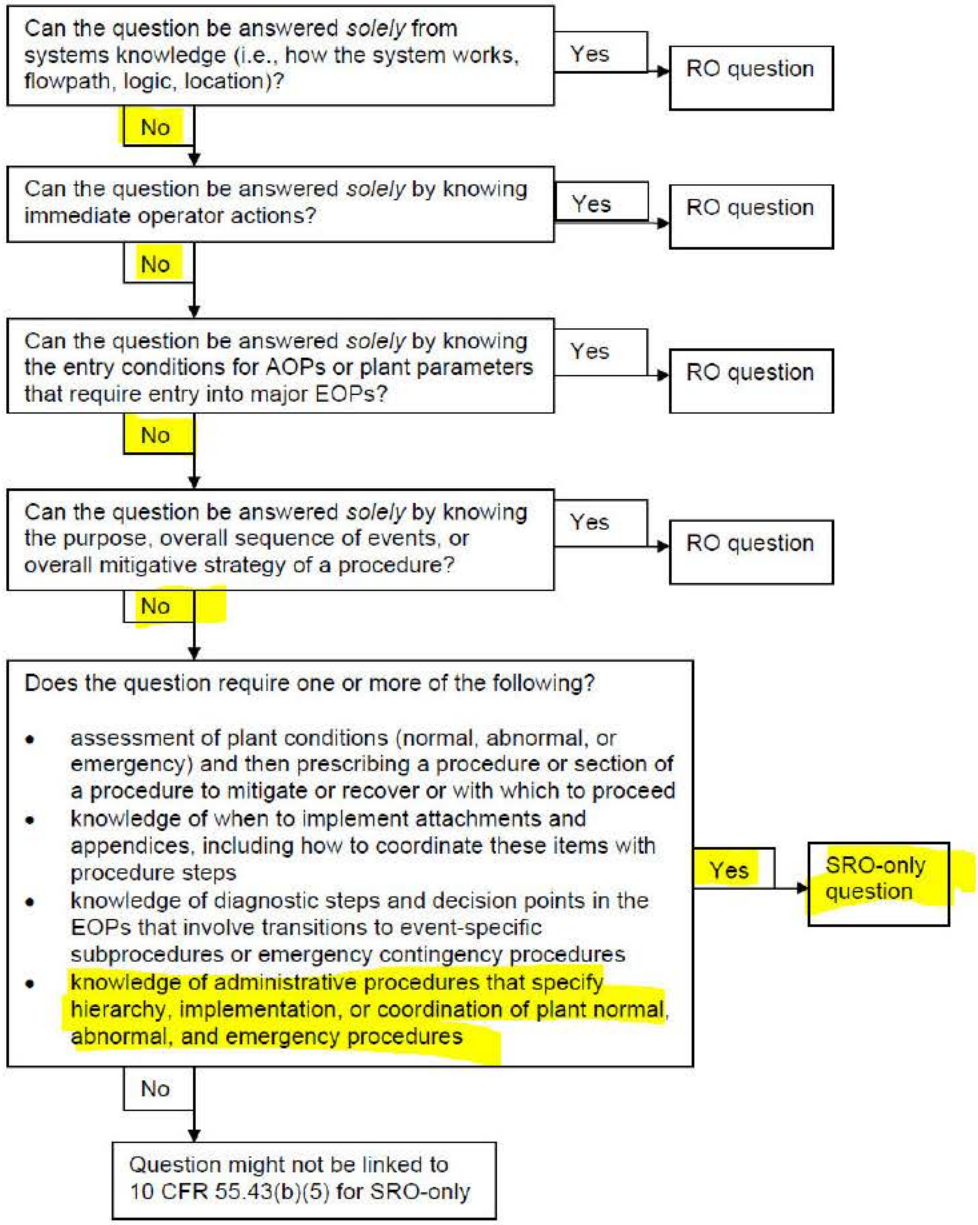
4.2.15. Status Tree monitoring should be continuous if any PURPLE or RED condition is found to exist. If a continuous RED or PURPLE priority is indicated, the performance of the appropriate CSF should be completed based on the priority rules of usage as discussed above. If no condition more serious than YELLOW is encountered, monitoring frequency may be reduced to 15 minutes, unless some significant change in plant status occurs.

4.2.16. A YELLOW terminus does not require immediate operator attention. Frequently it is indicative of an off-normal and/or temporary condition which will be restored to normal status by actions already in progress. In other cases, the YELLOW status might provide an early indication of a developing RED or PURPLE condition. Following FRG implementation, a YELLOW might indicate a residual off-normal condition. The operator is allowed to decide whether or not to implement any YELLOW condition FRP.

4.2.17. Implementation of a yellow path function restoration procedure is based on operator judgement when it is determined that adequate time exists to implement it. In other words, the operator does not have to implement a yellow path procedure if a judgement has been made that it is inappropriate based on available time or current plant state; and, if an event of higher priority is in progress, the operator should attend to the more important matters prior to implementing a yellow path function restoration procedure.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



100 **15-045** **Points: 1.00**

Redacted Due to Security-Related Information

Given:

- A truck accident has rendered BOTH Salem Fire Pumps INOPERABLE
- The crew is implementing S2.OP-AB.FP-0001, Fire Protection System Malfunction

In accordance with S2.OP-AB.FP-0001, which of the following completes both statements?

1. What action will the CRS direct?

2. If the action cannot be established within 24 hours, then the unit must be in _____ within the next 6 hours?

A. (1) Open the backup supply cross-tie valves from Hope Creek.
(2) HOT STANDBY

B. (1) Open the backup supply cross-tie valves from Hope Creek.
(2) HOT SHUTDOWN

C. (1) [redacted]
(2) HOT SHUTDOWN

D. (1) [redacted]
(2) HOT STANDBY

Redacted Due to Security-Related Information

Answer: A **Redacted Due to Security-Related Information**

Explanation / Justification:

A. Correct. Per S2.OP-AB.FIRE-0002, step 3.4 if at least one fire pump and one fire pump water supply is NOT available, then step 3.5 will direct opening the backup fire suppression water supply cross-tie valves from Hope Creek. If this cannot be established within 24 hours (step 3.6) then a unit shutdown is required and Hot Standby within the next 6 hours, Hot Shutdown within the following 6 hours, and Cold Shutdown within the subsequent 24 hours.

B. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because Hot Shutdown is a required MODE, but in the next 6 hours after reaching Hot Standby in 6 hours.

C. Incorrect. Part 1 is incorrect. Plausible because [redacted], but not directed by procedure. Part 2 is incorrect. Plausible because Hot Shutdown is a required MODE, but in the next 6 hours after reaching Hot Standby in 6 hours.

D. Incorrect. Part 1 is incorrect. Plausible because [redacted], but not directed by procedure. Part 2 is correct.

Technical References:	S2.OP-AB.FIRE-0002 (R2)
Proposed References to be provided:	None
Learning Objective:	NOS05ABFP01-06, Objective 2. Describe, in general terms, the actions taken in S2.OP-AB.FP-0001 and the bases for the actions.
Question Source:	Bank - Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	SRO
Category/System:	15-Emergency Procedures/Plan
K/A #:	G2.4.25 Knowledge of fire protection procedures.
Importance:	3.7
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the SRO is being asked the required action for when both Salem Fire Pumps are inoperable and the plant impact if the cross-tie cannot be opened within 24 hours.

Redacted Due to Security-Related Information

Additional Reference:

S2.OP-AB.FP-0001(Q)

___ 3.5 **NOTIFY** Site Protection to establish a Backup Fire Suppression Water System supply from Hope Creek, **OPEN** cross tie valves:

- 1PP30, FP HDR TO HOPE CREEK IV
- 0-KC-V115, HOPE CREEK/SALEM CROSSCONNECT VALVE

3.6 Can a backup Fire Suppression Water System supply be established within 24 hours?

___ NO ___ YES ——> **GO TO** Step 3.9 _____ Time

↓

___ 3.7 **NOTIFY** Operations Management of the unavailability of BOTH Fire Protection Water Systems.

___ 3.8 **INITIATE** Unit Shutdown IAW the appropriate IOP as follows: **[C0657]**

___ A. Be in at least HOT STANDBY within the next 6 hours.

___ B. Be in at least HOT SHUTDOWN within the following 6 hours.

___ C. Be in at least COLD SHUTDOWN within the subsequent 24 hours.

SRO Match Justification

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)