EPE007 EK1.04 - Reactor Trip

Knowledge of the operational implications of the following concepts as they apply to the reactor trip: (CFR 41.8 / 41.10 / 45.3)

Decrease in reactor power following reactor trip (prompt drop and subsequent decay) ....

Given the following Unit 1 conditions:

- Reactor trip has just occurred from 100% power
- Reactor power is less than 1 percent and decreasing

1) As the control rods are inserting from the trip, startup rate will be____(1)____ before stabilizing at – 1/3 DPM while power decreases towards the source range.

2) If ONE control rod remains fully withdrawn after the reactor trip, the EOP ____(2)____ direct boration from the BWST.

Which ONE of the following completes the statements above?

A. 1. greater than -1/3 DPM (more negative)
   2. does

B. 1. greater than -1/3 DPM (more negative)
   2. does NOT

C. 1. less than -1/3 DPM (less negative)
   2. does

D. 1. less than -1/3 DPM (less negative)
   2. does NOT

General Discussion

Need to ensure that with new RPS/ES that a single CR can not still be energized with the rest of the controls inserted from a trip.

Answer A Discussion

1st part is correct. When a trip occurs, the prompt drop in neutron populaton will result in a negative startup rate that exceeds - 1/3 DPM. As the short lived DNPs decay away, SUR will become a stable ~ -1/3 DPM as power decreases towards the source range.

2nd part is correct. Per subsequent actions, if ALL control rods are not inserted (Ex Gp 8), SA directs the operator to open 1 HP-24 and 1HP-25. This is to ensure that 1% SDM is maintained.

Answer B Discussion

1st part is correct.

2nd part is plausible since the reactor is shutting down and is below 1% power and the Shutdown Margin curves always assume the worst case control rod remains withdrawn therefore it would be logical to deduce that based on the indications given in the stem opening 1HP-24 and 1HP-25 (HPI suction to the BWST) would not be required.

Answer C Discussion

1st part is incorrect because the initial startup rate on a trip would be greater than -1/3 DPM. It is plausible because it is a common misconception that -1/3 DPM is the maximum negative startup rate than you can achieve on a reactor trip.

2nd part is correct. Per subsequent actions, if ALL control rods are not inserted (Ex Gp 8), SA directs the operator to open 1 HP-24 and 1HP-25. This is to ensure that 1% SDM is maintained.

Answer D Discussion

1st part is incorrect because the initial startup rate on a trip would be greater than -1/3 DPM. It is plausible because it is a common misconception that -1/3 DPM is the maximum negative startup rate than you can achieve on a reactor trip.

2nd part is plausible since the reactor is shutting down and is below 1% power and the Shutdown Margin curves always assume the worst case control rod remains withdrawn therefore it would be logical to deduce that based on the indications given in the stem opening 1HP-24 and 1HP-25 (HPI suction to the BWST) would not be required.

Basis for meeting the KA

This question matches the KA by requiring knowledge of how power decreases on a reactor trip and how operator actions are different with a single control rod stuck out.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References

Rx Op Physics, Obj: 26, pg 33
EOP SA
EAP-SA Obj: R1

Student References Provided

EPE007 EK1.04 - Reactor Trip

Knowledge of the operational implications of the following concepts as they apply to the reactor trip: (CFR 41.8 / 41.10 / 45.3)

Decrease in reactor power following reactor trip (prompt drop and subsequent decay) ....

401-9 Comments:

Remarks/Status

APE008 2.4.31 - Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)
APE008 GENERIC
Knowledge of annunciator alarms, indications, or response procedures. (CFR: 41.10 / 45.3)

Unit 1 plant conditions:

- Reactor Power =100%
- 1SA-18, A/1 PRESSURIZER RELIEF VALVE FLOW alarms
- RCS pressure = 2200 psig decreasing
- 1RC-66 indicates partially open
- 1RC-4 will not close from the control room

____(1)____ will be entered which will dispatch an operator to open 1DIB Breaker # 24 to fail ____(2)____ closed.

Which ONE of the following completes the statement above?

A. 1. AP/2, (Excessive RCS Leakage)
   2. 1RC-66

B. 1. AP/2, (Excessive RCS Leakage)
   2. 1RC-4

C. 1. AP/44, (Abnormal Pressurizer Pressure Control)
   2. 1RC-66

D. 1. AP/44, (Abnormal Pressurizer Pressure Control)
   2. 1RC-4

General Discussion

Answer A Discussion

1st part is incorrect because there is no direction in AP/2 to open the breaker for 1RC-66. It is plausible because 1) you meet entry conditions for AP/2, 2) AP/44 directs entry into AP/2 and AP/2 Encl 5.9 does give direction to close 1RC-4 if leakage through 1RC-66 exceeds 1 gpm.

2nd part is correct. AP/44, Step 4.3 RNO directs opening the breaker for 1RC-66. The PORV will fail closed (unless mechanically stuck) when power is removed.

Answer B Discussion

1st part is incorrect because there is no direction in AP/2 to open the breaker for 1RC-66. It is plausible because 1) you meet entry conditions for AP/2, 2) AP/44 directs entry into AP/2 and AP/2 Encl 5.9 does give direction to close 1RC-4 if leakage through 1RC-66 exceeds 1 gpm.

2nd part is incorrect because bkr # 24 is the power supply to 1RC-66. 1RC-4 is an MOV so it will fail as is. It is plausible because this is the RNO step for 1RC-4 failing to close from the control room.

Answer C Discussion

1st part is correct. AP/44 entry conditions are met. Step 4.3 RNO dispatches an operator to open the breaker for 1RC-66.

2nd part is correct. AP/44, Step 4.3 RNO directs opening the breaker for 1RC-66. The PORV will fail closed (unless mechanically stuck) when power is removed.

Answer D Discussion

1st part is correct. AP/44 entry conditions are met. Step 4.3 RNO dispatches an operator to open the breaker for 1RC-66.

2nd part is incorrect because bkr # 24 is the power supply to 1RC-66. 1RC-4 is an MOV so it will fail as is. It is plausible because this is the RNO step for 1RC-4 failing to close from the control room.

Basis for meeting the KA

This question matches the KA by requiring knowledge of the response procedure (AP/44) for a failed open relief valve.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

AP/44
OMP1-18
EAP-APG Obj: R9
EAP-APG 44
AP/2

Student References Provided

APE008 2.4.31 - Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)
APE008 GENERIC
Knowledge of annunciator alarms, indications, or response procedures. (CFR: 41.10 / 45.3)

401-9 Comments:

Remarks/Status

EPE009 EK2.03 - Small Break LOCA
Knowledge of the interrelations between the small break LOCA and the following: (CFR 41.7 / 45.7)
S/Gs ....

Given the following Unit 1 conditions:

Time = 0400
- Reactor Trip
- SBLOCA has occurred
- Rule 2 (Loss of SCM) in progress
- 1A and 1B MD EFDWPs are operating

Time = 0410
- Rule 7 (SG Feed Control) in progress
- RCS cooldown rate = 61°F/½ hr
- EFW flow = 100 gpm to each SG
- 1A and 1B SG levels = 85” XSUR stable

1) At 0400, in accordance with Rule 2, __ (1) __ gpm EFDW flow will initially be established to each SG.

2) At 0410, in accordance with Rule 7 (SG Feed Control), EFDW flow should be __ (2) __.

Which ONE of the following completes the statements above?

A. 1. 300
2. decreased

B. 1. 300
2. increased

C. 1. 450
2. decreased

D. 1. 450
2. increased

ILT 47 ONS SRO NRC Examination **QUESTION** **3** 3 **B**

General Discussion

Answer A Discussion

1st part is correct. Per Rule 2, step 42, Establish 300 gpm EFDW flow to each SG.

2nd part is incorrect because EFDW flow should be increased. Rule 2, note prior to step 47 states: SG levels must continue to increase until the SG level control point is reached. It is plausible because the TS cooldown rate is being exceeded at 0410. If you were not feeding to the LOSCM setpoint (not in Rule 2), it would be correct.

Answer B Discussion

1st part is correct. EFDW is initially set to 300 gpm to each SG per Rule 2.

2nd part is correct. Rule 2, note prior to step 47 states: SG levels must continue to increase until the SG level control point (LOSCM setpt) is reached. Even though the TS cooldown rate is currently being exceeded, SG level is NOT increasing towards the LOSCM setpt therefore, per Rule 2 guidance, flow must be increased to increase SG levels towards the LOSCM setpt.

Answer C Discussion

1st part is incorrect because Rule 2 direction is to feed SGs at 300 gpm each. It is plausible because if only one SG were available, it would be correct.

2nd part is incorrect because EFDW flow should be increased. Rule 2, note prior to step 47 states: SG levels must continue to increase until the SG level control point is reached. It is plausible because the TS cooldown rate is being exceeded at 0410. If you were not feeding to the LOSCM setpoint (not in Rule 2), it would be correct.

Answer D Discussion

1st part is incorrect because Rule 2 direction is to feed SGs at 300 gpm each. It is plausible because if only one SG were available, it would be correct.

2nd part is correct. Rule 2, note prior to step 47 states: SG levels must continue to increase until the SG level control point (LOSCM setpt) is reached. Even though the TS cooldown rate is currently being exceeded, SG level is NOT increasing towards the LOSCM setpt therefore, per Rule 2 guidance, flow must be increased to increase SG levels towards the LOSCM setpt.

Basis for meeting the KA

Question requires knowledge of how EFDW flow is established to SGs during a SBLOCA.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT41 (Q 2) NRC Exam

Development References

Rule 2
Rule 7
EAP LOSCM
ILT41 Q2

Student References Provided

EPE009 EK2.03 - Small Break LOCA
Knowledge of the interrelations between the small break LOCA and the following: (CFR 41.7 / 45.7)
S/Gs ....

401-9 Comments:

Remarks/Status

EPE011 EK2.02 - Large Break LOCA
Knowledge of the interrelations between the Large Break LOCA and the following: (CFR 41.7 / 45.7)
Pumps ....

Given the following Unit 1 conditions:
0800:

- Reactor power = 100%
- LBLOCA occurs

0810:

- RCS Pressure = 200 psig decreasing
- HPI Flow in 1A Header = 750 gpm
- HPI Flow in 1B Header = 490 gpm

Which ONE of the following describes the required operator actions to protect the HPI pumps?

A. Throttle HPI flows in BOTH 1A & 1B headers to <475 gpm per pump

B. Throttle HPI flow in ONLY 1A header to <750 gpm

C. Throttle HPI flows in BOTH 1A & 1B headers to <950 gpm combined

D. Throttle HPI flow in ONLY 1B header to <475 gpm

ILT 47 ONS SRO NRC Examination QUESTION 4 4 **D**

General Discussion

Answer A Discussion

Incorrect. Flow is acceptable in the A header due to 2 pumps operating aligned to the header. B Header flow requires throttling to <475 gpm per Rule 6. Plausible as the direction is correct if only one HPI pump is supplying the A header.

Answer B Discussion

Incorrect. Plausible as this is the value of total flow in Rule 6 when operating HPI in piggyback mode with either only one LPI pump running or only one piggyback valve open.

Answer C Discussion

Incorrect: Plausible as this is the value in Rule 6 if only HPI A & B operating with HP-409 open.

Answer D Discussion

Correct: Flow is above 475 flow limit and throttling is required per Rule 6

Basis for meeting the KA

Requires knowledge of relationship between HPI pump status and flow to determine required HPI pump throttling criteria to ensure pump operation within limits and core cooling is maintained.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	2009 (Q 4) NRC Exam

Development References

Rule 6
EAP-EOP Obj R 27
2009 Q4

Student References Provided

EPE011 EK2.02 - Large Break LOCA

Knowledge of the interrelations between the Large Break LOCA and the following: (CFR 41.7 / 45.7)

Pumps ....

401-9 Comments:

Remarks/Status

APE015/017 AK2.07 - Reactor Coolant Pump (RCP) Malfunctions
Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: (CFR 41.7 / 45.7)
RCP seals ....

Given the following Unit 1 conditions:
Initial Conditions

- Core Thermal Power = 100%.

Current conditions:

- A Station Blackout occurs at 0600.
- AP/0/A/1700/025 (Standby Shutdown Facility Emergency Operating Procedure) has been initiated.
- 1XSF is being powered from 0XSF.

1) In accordance with station Time Critical Actions, SSF RCMU flow must be established to Unit 1 RCP seals no later than ___(1)___.

2) 1HP-20 (RCP Seal Return) ___(2)__ be operated from Unit 1 Control Room at this time.

Which ONE of the following completes the statements above?

A. 1. 0614
2. can

B. 1. 0620
2. can

C. 1. 0614
2. cannot

D. 1. 0620
2. cannot

ILT 47 ONS SRO NRC Examination QUESTION 5 | 5 | D

General Discussion

Answer A Discussion

1st part is incorrect because it is time critical that flow be established to RCP seals within 20 minutes. It is plausible because for the same event that SSF-ASW be established to the SGs within 14 minutes.

2nd part is incorrect because 1HP-20 can not be operated from the control room once power has been switched to the SSF. It is plausible because it normally is powered from the control room and not all valves transfer control to the SSF.

Answer B Discussion

1st part is correct. Establishing RCP seals is required to be established within 20 minutes.

2nd part is incorrect because 1HP-20 can not be operated from the control room once power has been switched to the SSF. It is plausible because it normally is powered from the control room and not all valves transfer control to the SSF.

Answer C Discussion

1st part is incorrect because it is time critical that flow be established to RCP seals within 20 minutes. It is plausible because for the same event that SSF-ASW be established to the SGs within 14 minutes.

2nd part is correct. 1HP-20 can not be operated from the control room once power has been switched to the SSF.

Answer D Discussion

1st part is correct. Establishing RCP seals is required to be established within 20 minutes.

2nd part is correct. 1HP-20 can not be operated from the control room once power has been switched to the SSF.

Basis for meeting the KA

Question matches the KA by requiring knowledge of the process for re-establishing RCP seals after RCPs are lost (due to blackout).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT43 (Q 5) NRC Exam

Development References

EAP-SSF Pg 10, 16, 27
AP 25
ILT43 Q5

Student References Provided

APE015/017 AK2.07 - Reactor Coolant Pump (RCP) Malfunctions

Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: (CFR 41.7 / 45.7)

RCP seals ....

401-9 Comments:

Remarks/Status

APE022 AK1.02 - Loss of Reactor Coolant Makeup

Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Makeup: (CFR 41.8 / 41.10 / 45.3)

Relationship of charging flow to pressure differential between charging and RCS . . . .

Given the following Unit 1 conditions:

Time = 1200

- Reactor power = 80%
- 1HP-120 (RC VOLUME CONTROL) FAILED CLOSED
- Makeup flow has been re-established in accordance with AP/14 (Loss of Normal HPI Makeup and/or RCP Seal Injection)

Time = 1215

- Pressurizer level is 220" stable

1) In accordance with AP/14, ____(1)____ was throttled first to maintain Pzr level.

2) If 1RC-1 subsequently fails open at Time = 1220, prior to any Operator actions RCS makeup flow will ____(2)____.

Which ONE of the following completes the statements above?

A. 1. 1HP-26 (1A HP INJECTION)
   2. increase

B. 1. 1HP-26 (1A HP INJECTION)
   2. decrease

C. 1. 1HP-122 (RC VOLUME CONTROL BYPASS)
   2. increase

D. 1. 1HP-122 (RC VOLUME CONTROL BYPASS)
   2. decrease

ILT 47 ONS SRO NRC Examination QUESTION 6

6

A

General Discussion

Answer A Discussion

1st part is correct. In AP/14, step 4.176, it directs maintaining Pzr level > 200" using 1HP-126.

2nd part is correct, using pump laws, whe RCS pressure decreases as a result of the failed open spray valve the dp between pump discharge and RCS pressure will increase therefore HPI pump flow will increase.

Answer B Discussion

1st part is correct. In AP/14, step 4.176, it directs maintaining Pzr level > 200" using 1HP-126.

2nd part is incorrect but plausible since makeup flow changes inversely with RCS pressure. Since the change in flow is proportional to the change in RCS pressure it would be easy to confuse the direction of change since it is the inverse of pressure.

Answer C Discussion

1st part is incorrect because AP/14 directs throttling 1HP-26 to maintain Pzr level. It is plausible because if 1HP-26 does not work, it does direct you to throttle 1HP-122.

2nd part is correct, using pump laws, whe RCS pressure decreases as a result of the failed open spray valve the dp between pump discharge and RCS pressure will increase therefore HPI pump flow will increase.

Answer D Discussion

1st part is incorrect because AP/14 directs throttling 1HP-26 to maintain Pzr level. It is plausible because if 1HP-26 does not work, it does direct you to throttle 1HP-12.

2nd part is incorrect but plausible since makeup flow changes inversely with RCS pressure. Since the change in flow is proportional to the change in RCS pressure it would be easy to confuse the direction of change since it is the inverse of pressure.

Basis for meeting the KA

The question matches the KA by requiring knowledge of how to determine charging flow rate based on changing the HPIP / RCS DP.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References
AP/14 Step 146 EAP-APG Obj R9

Student References Provided

APE022 AK1.02 - Loss of Reactor Coolant Makeup

Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Makeup: (CFR 41.8 / 41.10 / 45.3)

Relationship of charging flow to pressure differential between charging and RCS . . . .

401-9 Comments:

Remarks/Status

ILT 47 ONS SRO NRC Examination QUESTION 7 7 **D**

APE026 AA2.06 - Loss of Component Cooling Water (CCW)

Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: (CFR: 43.5 / 45.13)

The length of time after the loss of CCW flow to a component before that component may be damaged ....

Given the following Unit 1 conditions

- Reactor power = 100%
- The running CC pump trips
- AP/20 LOSS OF COMPONENT COOLING has been entered
- 2 minutes have elapsed
- CC Surge Tank level = 10”
- NO automatic actions occur

1) Assuming that no operator actions are taken during the first 2 minutes of the event, ____(1)____.

2) Based on the above plant conditions, AP/20 ____(2)____ direct the operator to start the standby CC pump.

Which ONE of the following completes the statements above?

A. 1. CRDM temperatures will have increased to the point at which damage has occurred to the stator windings
   2. will

B. 1. CRDM temperatures will have increased to the point at which damage has occurred to the stator windings
   2. will NOT

C. 1. Demineralizer temperatures will have increased to the point at which damage has occurred to the demineralizer resin
   2. will

D. 1. Demineralizer temperatures will have increased to the point at which damage has occurred to the demineralizer resin
   2. will NOT

ILT 47 ONS SRO NRC Examination QUESTION 7 7 **D**

General Discussion

Answer A Discussion

1st part is incorrect because the time given in a Caution contained in AP/20 is that in ~ 4 minutes, CRDM temperature will exceed 180 degrees. When 2 CRDMs reach that temperature, it directs tripping of the reactor. It is plausible because if it were > 4 minutes, it may be correct. 2nd part is incorrect because in AP/20 a pre-requisite to starting the standby CC pump is that CC surge tank is > 12 ". It is plausible because if ST level were > 12", it would be correct.

Answer B Discussion

1st part is incorrect because the time given in a Caution contained in AP/20 is that in ~ 4 minutes, CRDM temperature will exceed 180 degrees. When 2 CRDMs reach that temperature, it directs tripping of the reactor. It is plausible because if it were > 4 minutes, it may be correct.

2nd part is correct. Per AP/20, CC Surge Tank level must be > 12" prior to starting the Standby CC pump.

Answer C Discussion

1st part is correct. Letdown should isolate at 130 degrees to prevent resin damage which will start to occur with the anion resin at ~ 140 degrees. This temperature would be reached within seconds of losing CC cooling to the letdown heat exchanger.

2nd part is incorrect because in AP/20 a pre-requisite to starting the standby CC pump is that CC surge tank is > 12 ". It is plausible because if ST level were > 12", it would be correct.

Answer D Discussion

1st part is correct. Letdown should isolate at 130 degrees to prevent resin damage which will start to occur with the anion resin at ~ 140 degrees. This temperature would be reached within seconds of losing CC cooling to the letdown heat exchanger.

2nd part is correct. Per AP/20, CC Surge Tank level must be > 12" prior to starting the Standby CC pump.

Basis for meeting the KA

This question matches the KA by requiring knowledge of how long it takes to cause component damage once Component Cooling water is lost.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References

AP 20
EAP APG Obj R9
BNT-CH05
PNS HPI Pg 15
ARG CRD Ret Flow Lo

Student References Provided

APE026 AA2.06 - Loss of Component Cooling Water (CCW)

Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: (CFR: 43.5 / 45.13)

The length of time after the loss of CCW flow to a component before that component may be damaged ....

401-9 Comments:

Remarks/Status

EPE029 2.4.18 - Anticipated Transient Without Scram (ATWS)
EPE029 GENERIC
Knowledge of the specific bases for EOPs. (CFR: 41.10 / 43.1 / 45.13)

Given the following Unit 1 conditions:
Initial conditions:

- Reactor power = 100%
- RCS pressure = 2360 psig increasing

Current conditions:

- Reactor power = 7% decreasing

1) With Reactor power decreasing, the MINIMUM power level at which Rule 1 (ATWS/UNPP) is required to be performed to address Emergency Boration is __(1)__.

2) The reason this power level is chosen is so the Boron will reduce reactor power to __ (2) __.

Which ONE of the following completes the statements above?

A. 1. 1%
2. below the point of adding heat

B. 1. 1%
2. within the capacity of the EFDW system

C. 1. 5%
2. below the point of adding heat

D. 1. 5%
2. to within the capacity of the EFDW system

General Discussion

Answer A Discussion

Incorrect. First part is plausible because HPI can be throttled below 1% power. Second part is plausible because of a misconception that power is reduce so that the no nuclear heat is being added to the system.

Answer B Discussion

Incorrect. First part is plausible because HPI can be throttled below 1% power. Second part is correct.

Answer C Discussion

Incorrect. First part is correct. Second part is plausible because of a misconception that power is reduce so that the no nuclear heat is being added to the system.

Answer D Discussion

Correct. During performance of IMAs, if power is greater than 5% Rule 1 must be performed. This is to reduce reactor power to within the heat removal capacity of the EFDW system.

Basis for meeting the KA

Question requires knowledge of the reason for the power level that will require emergency boration to be performed.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT41 (Q 19) NRC Exam

Development References

EOP IMAs
EOP RULE 1
EAP-UNPP Pg 7
ILT41 Q19

Student References Provided

EPE029 2.4.18 - Anticipated Transient Without Scram (ATWS)
EPE029 GENERIC
Knowledge of the specific bases for EOPs. (CFR: 41.10 / 43.1 / 45.13)

401-9 Comments:

Remarks/Status

EPE038 EK3.01 - Steam Generator Tube Rupture (SGTR)
Knowledge of the reasons for the following responses as the apply to the SGTR: (CFR 41.5 / 41.10 / 45.6 / 45.13)
Equalizing pressure on primary and secondary sides of ruptured S/G ......

Given the following Unit 1 conditions:

- SGTR tab in progress
- 1B SG isolated
- 1B1 RCP secured
- 1A loop Tcold = 440°F decreasing
- 1B S/G TUBE/SHELL DT = (-)72°F

1) The reason the SGTR tab directs minimizing core SCM during cooldown is to minimize__(1)__.

2) The initial method that will be used to reduce the SCM is __(2)__.

Which ONE of the following completes the statements above?

A. 1. primary to secondary leak rate
2. de-energizing Pzr heaters and cycling Pzr spray

B. 1. primary to secondary leak rate
2. cycling the PORV

C. 1. compressive stresses in the 1B SG
2. de-energizing Pzr heaters and cycling Pzr spray

D. 1. compressive stresses in the 1B SG
2. cycling the PORV

ILT 47 ONS SRO NRC Examination QUESTION 9

9

A

General Discussion

Answer A Discussion

CORRECT: The purpose of reducing SCM during a SGTR is to reduce RCS pressure as much as possible while still maintaining SCM and RCP NPSH. This minimizes the differential pressure between the RCS and the affected SG(s), thus minimizing the tube leak flow rate. The SGTR tab directs the operator to initially use pressurizer heaters and normal Pzr spray. If initial methods do not achieve desired results the PORV is cycled to reduce the SCM.

Answer B Discussion

Incorrect: First part is correct. Second part is plausible since the 1B1 RCP has been secured and on unit 3 that is the RCP in the Pzr spray loop. Using the PORV is a strategy used in the SGTR tab to reduce SCM however it is not used unless initial methods attempted are inadequate.

Answer C Discussion

Incorrect: First part is plausible since controlling compressive stresses across SG tubes is a prime concern during SGTR. 1B Tube/Shell delta T is violating the Compressive stress limit of -70°F. However, reducing SCM is not a strategy directed at correcting this issue. Feeding the isolated SG would be used to reduce the Compressive stresses. Second part is correct.

Answer D Discussion

Incorrect: First part is plausible since controlling compressive stresses across SG tubes is a prime concern during SGTR. 1B Tube/Shell delta T is violating the Compressive stress limit of -70°F. However, reducing SCM is not a strategy directed at correcting this issue. Feeding the isolated SG would be used to reduce the Compressive stresses.

Second part is plausible since the 1B1 RCP has been secured and on unit 3 that is the RCP in the Pzr spray loop. Using the PORV is a strategy used in the SGTR tab to reduce SCM however it is not used unless initial methods attempted are inadequate.

Basis for meeting the KA

Requires knowing the reason for equalizing pressure on primary and secondary sides of ruptured S/G and how that is done.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	2010A (Q 9) NRC Exam

Development References

EAP-SGTR Steps 38-42
EOP reference document
EAP-SGTR
2010A Q9

Student References Provided

EPE038 EK3.01 - Steam Generator Tube Rupture (SGTR)

Knowledge of the reasons for the following responses as the apply to the SGTR: (CFR 41.5 / 41.10 / 45.6 / 45.13)

Equalizing pressure on primary and secondary sides of ruptured S/G ......

401-9 Comments:

Remarks/Status

APE040 AA1.18 - Steam Line Rupture
Ability to operate and / or monitor the following as they apply to the Steam Line Rupture: (CFR 41.7 / 45.5 / 45.6)
Control rod position indicators ....

Given the following Unit 1 conditions:
Time = 0800:00

- Reactor power = 90%
- Control Rod Gp 7 position = 90%
- The PI panel is selected to "Relative"
- Steam line break on the 1A SG occurs inside containment

Time = 0801:00

- Reactor trip occurs

1) At Time = 0800:30 ____(1)____ SG pressure(s) will be decreasing.

2) The MINIMUM requirement for Relative Position Indication (RPI) to AUTOMATICALLY reset to 0% is to have ____(2)____.

Which ONE of the following completes the statements above?

A. 1. ONLY 1A
2. a Trip Confirmed signal generated

B. 1. ONLY 1A
2. ALL Regulating Rod Group IN LIMITs satisfied

C. 1. BOTH 1A and 1B
2. a Trip Confirmed signal generated

D. 1. BOTH 1A and 1B
2. ALL Regulating Rod Group IN LIMITs satisfied

ILT 47 ONS SRO NRC Examination QUESTION 10

10

C

General Discussion

Answer A Discussion

1st part is incorrect because prior to the reactor trip, both SGs pressures will be decreasing. It is plausible because after the reactor trip, it would be correct..

2nd part is correct. The trip confirm signal resets the RPI to the API position which should be on the bottom or RPI ~ 0%.

Answer B Discussion

1st part is incorrect because prior to the reactor trip, both SGs pressures will be decreasing. It is plausible because after the reactor trip, it would be correct..

2nd part is incorrect but plauible since the group in limit lights are fed from API indiciation which would indicate actual rod position therefore the group in limit light would be a good indication that the group of rods are fully inserted and therefore a reasonable choice of indiciation to use to reset RPI. Also, there is an auto action that is initiated by the group in limit therefore it would be an easy misconception that it was used to reset RPI indiciations.

Answer C Discussion

1st part is correct Until the Turbine Stop Valves close, the steam pressure in both headers will be decreasing.

2nd part is correct. The trip confirm signal resets the RPI to the API position which should be on the bottom or RPI ~ 0%.

Answer D Discussion

1st part is correct Until the Turbine Stop Valves close, the steam pressure in both headers will be decreasing.

2nd part is incorrect but plauible since the group in limit lights are fed from API indiciation which would indicate actual rod position therefore the group in limit light would be a good indication that the group of rods are fully inserted and therefore a reasonable choice of indiciation to use to reset RPI. Also, there is an auto action that is initiated by the group in limit therefore it would be an easy misconception that it was used to reset RPI indiciations.

Basis for meeting the KA

This question matches the KA by requiring knowledge of control rod position indicators react upon a trip.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References

IC-CRI Pg 22, 23
IC-RPS
STG-MS Pg 11

Student References Provided

APE040 AA1.18 - Steam Line Rupture

Ability to operate and / or monitor the following as they apply to the Steam Line Rupture: (CFR 41.7 / 45.5 / 45.6)

Control rod position indicators ....

401-9 Comments:

Remarks/Status

APE054 AA2.06 - Loss of Main Feedwater (MFW)
Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): (CFR: 43.5 / 45.13)
AFW adjustments needed to maintain proper T-ave. and S/G level ....

Given the following Unit 1 conditions:
Initial conditions:

- Reactor power = 100%
- Both Main FDW pumps trip
- 1A and 1B MDEFDW pumps did NOT start
- TDEFWP did NOT start

Current conditions:

- Tave = 566ºF stable
- Recovery from CBP feed with the TDEFDW pump is in progress
- TDEFWP is running and flow has been verified

Which ONE of the following describes how Tave and SG levels will be controlled INITIALLY during the recovery from CBP feed?

Tave will INITIALLY be controlled by throttling ____(1)____ and INITIALLY a SG level ____(2)____ be established.

A. 1. EFDW flow
2. will NOT

B. 1. the TBVs
2. will NOT

C. 1. EFDW flow
2. will

D. 1. the TBVs
2. will

General Discussion

Answer A Discussion

1st part is correct. Per the LOHT tab, if Tcold > 547 degrees, The THP setpoint is set at 885 psig. Feed is then initiated to establish heat transfer.

2nd part is correct. SG level will not be established at this point because if it were, Tc would rapidly decrease to whatever Tsat is for SG pressure. This rapid cooldown is not desired.

Answer B Discussion

1st part is incorrect because EFW will be used to control Tave. It is plausible because if at this point in the LOHT tab and Tc was < 547 degrees, it would be correct.

2nd part is correct. SG level will not be established at this point because if it were, Tc would rapidly decrease to whatever Tsat is for SG pressure. This rapid cooldown is not desired.

Answer C Discussion

1st part is correct. Per the LOHT tab, if Tcold > 547 degrees, The THP setpoint is set at 885 psig. Feed is then initiated to establish heat transfer.

2nd part is incorrect because a level will not be established at this point. It is plausible because it would be correct if Tave was less than 547 degrees which would allow establishing a SG level without excessive cooldown

Answer D Discussion

1st part is incorrect because EFW will be used to control Tave. It is plausible because if at this point in the LOHT tab and Tc was < 547 degrees, it would be correct.

2nd part is incorrect because a level will not be established at this point. It is plausible because it would be correct if Tave was less than 547 degrees which would allow establishing a SG level without excessive cooldown

Basis for meeting the KA

This question matches the KA by requiring the ability to intepret plant conditions and based on that interpretation, adjust AFW as required for Tave and SG level control.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	2009B Q11 NRC Exam

Development References
2009B Q11 LOHT Tab EAP-LOHT Pg 12

Student References Provided

APE054 AA2.06 - Loss of Main Feedwater (MFW)

Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): (CFR: 43.5 / 45.13)

AFW adjustments needed to maintain proper T-ave. and S/G level ....

401-9 Comments:	Remarks/Status

APE056 AA1.05 - Loss of Offsite Power

Ability to operate and / or monitor the following as they apply to the Loss of Offsite Power: (CFR 41.7 / 45.5 / 45.6)

Initiation (manual) of safety injection process ....

Given the following Unit 1 conditions:

Initial Conditions:

- Reactor power = 100%
- 1A HPI pump switch in ON
- 1B HPI pump switch in AUTO
- SBLOCA occurs
- The reactor trips on variable low RCS pressure

Current conditions

- 1A HPI pump switch in ON
- 1B HPI pump switch in AUTO
- A Switchyard Isolation occurs
- CT-1 locks out

1) Based on the Initial Conditions, as RCS pressure decreases, the operator ____(2)____ expected to manually initiate ES 1 & 2 prior to reaching the Emergency Injection setpoint.

2) Following the CT-1 lockout, when the 4160 VAC busses re-energize, there will be ____(1)____ HPIP(s) operating. (Assuming RCS pressure has decreased to 1500 psig while power was lost)

Which ONE of the following completes the statements above?

A. 1. is
   2. 0

B. 1. is
   2. 3

C. 1. is NOT
   2. 0

D. 1. is NOT
   2. 3

ILT 47 ONS SRO NRC Examination QUESTION 12

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B

General Discussion

Answer A Discussion

1st part is correct. AD-OP-ALL-1000, section 5.2.2 states: If degrading plant conditions are recognized in sufficient time, crews are expected to take manual actions prior to reaching the automatic setpoint for prescribed ESF and RPS actuations, unless otherwise directed by site specific procedures.

2nd part is incorrect because all 3 HPIPs will automaticallty start with the ES signal. It is plausible because not all ES related pumps will automaticallty start when power is regained (with an ES signal present) (C LPIP).

Answer B Discussion

1st part is correct. AD-OP-ALL-1000, section 5.2.2 states: If degrading plant conditions are recognized in sufficient time, crews are expected to take manual actions prior to reaching the automatic setpoint for prescribed ESF and RPS actuations, unless otherwise directed by site specific procedures

2nd part is correct. With an ES signal present, a start signal to all 3 HPIPs will be generated. When power is regained, all 3 will start.

Answer C Discussion

1st part is incorrect. AD-OP-ALL-1000, section 5.2.2 states: If degrading plant conditions are recognized in sufficient time, crews are expected to take manual actions prior to reaching the automatic setpoint for prescribed ESF and RPS actuations, unless otherwise directed by site specific procedures. It is plausible because there is OMP1-18 (Implementation Standard During Abnormal and Emergency Events) states that that states that manual actuation should be performed if automatic actuation setpoints have been reached and auto actuation has not occurred.

2nd part is incorrect because all 3 HPIPs will automaticallty start with the ES signal. It is plausible because not all ES related pumps will automaticallty start when power is regained (with an ES signal present) (C LPIP).

Answer D Discussion

1st part is incorrect. AD-OP-ALL-1000, section 5.2.2 states: If degrading plant conditions are recognized in sufficient time, crews are expected to take manual actions prior to reaching the automatic setpoint for prescribed ESF and RPS actuations, unless otherwise directed by site specific procedures. It is plausible because there is OMP1-18 (Implementation Standard During Abnormal and Emergency Events) states that that states that manual actuation should be performed if automatic actuation setpoints have been reached and auto actuation has not occurred.

2nd part is correct. With an ES signal present, a start signal to all 3 HPIPs will be generated. When power is regained, all 3 will start.

Basis for meeting the KA

This question matches the KA by requiring knowledge of manual starting of HPIPs during a loss of offsite power event (with a SB LOCA).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

OMP 1-18
AD-OP-ALL-1000
PNS HPI Pg 30
EL EPSL

Student References Provided

APE056 AA1.05 - Loss of Offsite Power

Ability to operate and / or monitor the following as they apply to the Loss of Offsite Power: (CFR 41.7 / 45.5 / 45.6)

Initiation (manual) of safety injection process ....

401-9 Comments:	Remarks/Status

APE057 AK3.01 - Loss of Vital AC Electrical Instrument Bus

Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: (CFR 41.5,41.10 / 45.6 / 45.13)

Actions contained in EOP for loss of vital ac electrical instrument bus ...

Given the following Unit 1 conditions:

Initial conditions:

- A loss of both MFW pumps occurs from 100% power
- Rule 3 (Loss of Main or Emergency FDW) is in progress
- 1FDW-315 and 1FDW-316 are maintaining SG levels at 30” XSUR

Current conditions:

- The breaker supplying power from KVIB to its associated SG level controller opens

Which ONE of the following will be directed by Rule 3 to control emergency feedwater flow and why?

A. Take manual control of 1FDW-315 and maintain level at 30” XSUR due to losing power for automatic control

B. Take manual control of 1FDW-316 and maintain level at 30” XSUR due to losing power for automatic control

C. Initiate Encl. 5.27 (Alternate Methods For Controlling EFDW Flow) and feed the 1A SG through 1FDW-35 (1A STARTUP FDW CONTROL) due to losing power to the Moore Controller

D. Initiate Encl. 5.27 (Alternate Methods For Controlling EFDW Flow) and feed the 1B SG through 1FDW-44 (1B STARTUP FDW CONTROL) due to losing power to the Moore Controller

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ILT 47 ONS SRO NRC Examination QUESTION 13 13 **A**

General Discussion

Answer A Discussion

Correct. When normal power is lost to the 1FDW-315 controller (KVIB), the controller (in automatic) will fail open. This will require the operator to place the controller in manual to control 1A SG level.

Answer B Discussion

Incorrect because KVIB is the normal power supply to the 1FDW-315 Controller. It is plausible because if it were KVIC, it would be correct.

Answer C Discussion

Incorrect because Rule 3 will direct taking manual control of 1FDW-315 which will be effective in controlling level. It is plausible because if the controller were not to work in manual either, it would be correct. KVIB is the power supply to the Moore controller but it automatically switches to KVIA upon power loss.

Answer D Discussion

Incorrect because KVIB is the normal power supply to the 1FDW-315 Controller. Plausible because if it were KVIC , it would apply to 1FDW-316. KVIC is the power supply to the Moore controller but it automatically switches to KVID upon power loss so it would still be incorrect if it were KVIC.

Basis for meeting the KA

This question matches the KA by requiring knowledge of how RO actions in Rule 3 (EOP) will be dictated by a loss of vital AC power (KVIB).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References
CF-EF Rule 3 Step 38-43

Student References Provided

APE057 AK3.01 - Loss of Vital AC Electrical Instrument Bus

Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: (CFR 41.5,41.10 / 45.6 / 45.13)

Actions contained in EOP for loss of vital ac electrical instrument bus ...

401-9 Comments:

Remarks/Status

APE058 AA1.03 - Loss of DC Power
Ability to operate and / or monitor the following as they apply to the Loss of DC Power: (CFR 41.7 / 45.5 / 45.6)
Vital and battery bus components ....

Given the following Plant conditions:

- 1CA Battery Charger fails - output voltage = 0 VDC
- 1CA Battery voltage = 126 VDC
- 1DCB Bus voltage = 123 VDC
- Unit 2 DCA/DCB Bus voltage = 124 VDC
- Unit 3 DCA/DCB Bus voltage = 127 VDC

Which ONE of the following will be supplying power to 1DIA panelboard?

A. 1DCB Bus

B. 1CA Battery

C. Unit 2 DC Bus

D. Unit 3 DC Bus

ILT 47 ONS SRO NRC Examination QUESTION 14

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B

General Discussion

Answer A Discussion

Incorrect. For the Vital DC system, the 1DCB bus is not aligned to the 1DCA bus. Plausible because 1DCB Bus is aligned to backup the essential inverters.

Answer B Discussion

Correct. The voltage from 1CA battery is higher than the backup source (Unit 2 DC Bus). Unit 1CA battery will supply power.

Answer C Discussion

Incorrect, plausible because this would be correct if the Unit 2 DC bus voltage was higher than the 1CA battery voltage.

Answer D Discussion

Incorrect. Unit 3's DC Bus is not connected to Unit 1. Plausible because unit 3 does backup Unit 1 in the SSF power scheme.

Basis for meeting the KA

Requires knowledge of the operational implications of failed battery charger and the operational impact of the loss of a Vital DC Battery Charger and the response by the Vital DC system

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	2010A (Q 13) NRC Exam

Development References	Student References Provided
EL-DCD Pg 22 2010A Q13	

APE058 AA1.03 - Loss of DC Power

Ability to operate and / or monitor the following as they apply to the Loss of DC Power: (CFR 41.7 / 45.5 / 45.6)

Vital and battery bus components ....

401-9 Comments:	Remarks/Status

APE062 2.2.42 - Loss of Nuclear Service Water
APE062 GENERIC
Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)

Given the following plant conditions:

0800:

- Unit 1 = 100% power
- Unit 2 = Mode 5
- Unit 3 = 100% power
- A, B and C LPSW pumps are operating

0805:

- C LPSW pump trips
- AP/24, Loss of LPSW is initiated
- LPSW header pressure = 65 psig stable

1) The LCO for TS 3.7.7 Low Pressure Service Water (LPSW) System ____(1)____ met for Unit 1.

2) Per AP/24, cross connecting Unit 1/2 LPSW system with Unit 3 LPSW ____(2)____ be directed.

Which ONE of the following completes the statements above?

A. 1. is
2. will

B. 1. is
2. will NOT

C. 1. is NOT
2. will

D. 1. is NOT
2. will NOT

General Discussion

Answer A Discussion

1st part is incorrect because the LCO for TS 3.7.7 is NOT met. It is plausible because there is a note in the spec that states only 2 LPSW pumps are required if one of the units is defueled (not the case in Mode 5). It Unit 2 was defueled, it could be correct.

2nd part is incorrect because cross connecting with Unit3 will only be performed if all Unit 1 & 2 LPSW is lost. It is plausible because LPSW pressure is still below the low pressure alarm setpoint with only two LPSW pumps operating.

Answer B Discussion

1st part is incorrect because the LCO for TS 3.7.7 is NOT met. It is plausible because there is a note in the spec that states only 2 LPSW pumps are required if one of the units is defueled (not the case in Mode 5). It Unit 2 was defueled, it could be correct.

2nd part iscorrect. Cross connecting with Unit3 will only be performed if all Unit 1 & 2 LPSW is lost which is not the case.

Answer C Discussion

1st part is correct. Since Unit 2 is in Mode 5, it still has fuel in it. Therefore, 3 LPSW pumps are required between Units 1 & 2 and the LCO is not met.

2nd part is incorrect because cross connecting with Unit3 will only be performed if all Unit 1 & 2 LPSW is lost. It is plausible because LPSW pressure is still below the low pressure alarm setpoint with only two LPSW pumps operating.

Answer D Discussion

1st part is correct. Since Unit 2 is in Mode 5, it still has fuel in it. Therefore, 3 LPSW pumps are required between Units 1 & 2 and the LCO is not met.

2nd part iscorrect. Cross connecting with Unit3 will only be performed if all Unit 1 & 2 LPSW is lost which is not the case.

Basis for meeting the KA

This question matches the KA by requiring knowledge of TS entry conditions for a loss of LPSW .

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

TS 3.7.7
SSS-LPW
AP/24

Student References Provided

APE062 2.2.42 - Loss of Nuclear Service Water
APE062 GENERIC
Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)

401-9 Comments:

Remarks/Status

APE065 AA2.08 - Loss of Instrument Air

Ability to determine and interpret the following as they apply to the Loss of Instrument Air: (CFR: 43.5 / 45.13)

Failure modes of air-operated equipment ....

Given the following Unit 1 conditions:

- Reactor power = 100%
- A complete loss of Instrument Air (IA) and Auxiliary Instrument Air (AIA) occurs.

Which ONE of the following describes RCP seal cooling and Pressurizer level response?

RCP Seal cooling will be maintained by ____(1)____ and pressurizer level will ____(2)____

ASSUME NO OPERATOR ACTIONS

A. 1. Component Cooling
2. decrease

B. 1. Component Cooling
2. increase

C. 1. HPI Seal Injection
2. decrease

D. 1. HPI Seal Injection
2. increase

ILT 47 ONS SRO NRC Examination QUESTION 16

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D

General Discussion

AP/22 requires the reactor to be tripped when FDW is not controllable. The OAC delta P can be expected at about 30 psig, well below the ~65 psig where FDW valves can stop responding to control signals. Applicants need to know when the OAC alarm actuates. Therefore, the AP requires that the reactor be tripped and the MFDW pumps to be tripped.

Answer A Discussion

1st part incorrect: 1CC-8, CC RETURN OUTSIDE BLOCK, fails closed, isolating the flowpath for RCP thermal barrier flow. It is plausible because some of the air valves have N2 backup.

Second part incorrect: 1HP-31, RCP SEAL FLOW CONTROL, fails open, providing more seal injection flow than before; 1HP-5, LETDOWN ISOLATION, fails closed. More flow into the RCS plus no flow out makes PZR level increase. It is plausible because 1HP-120 fails closed so the only makup is through 1HP-31. If 1HP-5 did not fail closed, it would be correct.

Answer B Discussion

1st part incorrect: 1CC-8, CC RETURN OUTSIDE BLOCK, fails closed, isolating the flowpath for RCP thermal barrier flow. It is plausible because some of the air valves have N2 backup.

2nd part is correct. Letdown (1HP-5 closed) has isolated and Makeup (1HP-120) has isolated. Seal injection (1HP-31) has failed open so the Pzr is filling from excess RCP seal injection.

Answer C Discussion

1st part is correct. CC-8 has failed closed which will stop CC flow to the seal cooler. Therefore seal injection is providing cooling to the seals.

Second part incorrect: 1HP-31, RCP SEAL FLOW CONTROL, fails open, providing more seal injection flow than before; 1HP-5, LETDOWN ISOLATION, fails closed. More flow into the RCS plus no flow out makes PZR level increase. It is plausible because 1HP-120 fails closed so the only makup is through 1HP-31. If 1HP-5 did not fail closed, it would be correct.

Answer D Discussion

1st part is correct. CC-8 has failed closed which will stop CC flow to the seal cooler. Therefore seal injection is providing cooling to the seals.

2nd part is correct. Letdown (1HP-5 closed) has isolated and Makeup (1HP-120) has isolated. Seal injection (1HP-31) has failed open so the Pzr is filling from excess RCP seal injection.

Basis for meeting the KA

This question matches the KA by requiring knowledge of the failure modes of air operated equiment that are supplied by IA.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT43 Q53

Development References

ILT43 Q53
AP/22 Encl 5.1
SSS IA Pg 48

Student References Provided

APE065 AA2.08 - Loss of Instrument Air

Ability to determine and interpret the following as they apply to the Loss of Instrument Air: (CFR: 43.5 / 45.13)

Failure modes of air-operated equipment ....

401-9 Comments:

Remarks/Status

D

ILT 47 ONS SRO NRC Examination QUESTION 17 17 **D**

BWE04 EK1.2 - Inadequate Heat Transfer

Knowledge of the operational implications of the following concepts as they apply to the (Inadequate Heat Transfer):

(CFR: 41.8 / 41.10 / 45.3)

Normal, abnormal and emergency operating procedures associated with (Inadequate Heat Transfer).

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%

Current conditions:

- Both Main Feedwater pumps tripped
- EFDW NOT available
- 1TD de-energized
- RCS pressure = 2217 psig slowly increasing

1) The pumps that will be aligned first to provide decay heat removal in accordance with the EOP are the __(1)__?

2) AP/11 (Recovery from Loss of Power) entry conditions __(2)__ met?

A.
1. HPI Pumps
2. are

B.
1. HPI Pumps
2. are NOT

C.
1. Condensate Booster Pumps
2. are

D.
1. Condensate Booster Pumps
2. are NOT

ILT 47 ONS SRO NRC Examination QUESTION 17

17

D

General Discussion

Answer A Discussion

1st part is incorrect because CBP feed will be initiated first. It is plausible because if RCS pressure were > 2300 psig, it would be correct. Additionally plausible since 1TD is de-energized therefore ALL CBP's are not available.

2nd part is incorrect It is plausible because AP/11 would be used if all of the three 4160V busses had been lost .

Answer B Discussion

1st part is incorrect because CBP feed will be initiated first. It is plausible because if RCS pressure were > 2300 psig, it would be correct. Additionally plausible since 1TD is de-energized therefore ALL CBP's are not available.

2nd part is correct. AP 11 is not entered unless all 3 4160v busses are lost.

Answer C Discussion

1st part is correct. With a loss of Main and EFDW, Rule 3 will establish CBP feed as long as it can be accomplished prior to reaching RCS pressure of 2300 psig. .

2nd part is incorrect It is plausible because AP/11 would be used if all of the three 4160V busses had been lost .

Answer D Discussion

1st part is correct. With a loss of Main and EFDW, Rule 3 will establish CBP feed as long as it can be accomplished prior to reaching RCS pressure of 2300 psig.

2nd part is correct. AP 11 is not entered unless all 3 4160v busses are lost.

Basis for meeting the KA

Requires knowledge of the operational implications of a loss of all Main and Emergency Feedwater in that knowledge of the procedural directed hierarchy of desired core cooling and the criteria for its use is required.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	2009B (Q17) NRC Exam

Development References

EAP-LOHT Pg 9
Rule 3
AP/11 Entry Conditions
2009B Q17

Student References Provided

BWE04 EK1.2 - Inadequate Heat Transfer

Knowledge of the operational implications of the following concepts as they apply to the (Inadequate Heat Transfer):
(CFR: 41.8 / 41.10 / 45.3)
Normal, abnormal and emergency operating procedures associated with (Inadequate Heat Transfer).

401-9 Comments:

Remarks/Status

APE077 AK3.01 - Generator Voltage and Electric Grid Disturbances

Knowledge of the reasons for the following responses as they apply to Generator Voltage and Electric Grid Disturbances: (CFR: 41.4, 41.5, 41.7, 41.10 / 45.8)

Reactor and turbine trip criteria…………......................................

Given the following Unit 1 conditions:
Time = 0800:

- Reactor power = 100%
- Grid voltage oscillating
- Generator Parameters:
  - MWe = 950
  - Vars = -350 MVAR
  - H2 Pressure = 45 psig
- AP/34 (Degraded Grid) has been initiated

Time = 0805:

- Reactor power = 60%
- Generator Parameters
  - MWe = 550
  - Vars = -450 MVAR
  - H2 Pressure = 45 psig
- It is determined that the Main Generator could sustain damage if it continued to operate

1) At 0805, AP/34 directs the operator to ____(1)____.

2) The reason this action is taken is to protect the generator from excessive ____(2)____.

Which ONE of the following competes the statements above?

REFERENCE PROVIDED

A. 1. OPEN PCB 20 and PCB 21
   2. armature core end heating

B. 1. OPEN PCB 20 and PCB 21
   2. field heating

C. 1. Trip the reactor
   2. armature core end heating

D. 1. Trip the reactor
   2. field heating

General Discussion

Answer A Discussion

1st part is incorrect because AP/34 directs tripping the reactor. It is plausible because if power were < 50%, it would be correct.

2nd part is correct per the Generator Capability Curve. Excessive Negative Vars/Vars /being underexcited can cause excessive armature core end heating.

Answer B Discussion

1st part is incorrect because AP/34 directs tripping the reactor. It is plausible because if power were < 50%, it would be correct.

2nd part is incorrect because underexciting the generator would result in armature core end heating.

Answer C Discussion

1st part is correct. With the stated conditions at 0805 including power being > 50%, AP/34 directs tripping the reactor.

2nd part is correct per the Generator Capability Curve. Excessive Negative Vars/Vars /being underexcited can cause excessive armature core end heating.

Answer D Discussion

1st part is correct. With the stated conditions at 0805 including power being > 50%, AP/34 directs tripping the reactor.

2nd part is incorrect because underexciting the generator would result in armature core end heating.

Basis for meeting the KA

This question matches the KA by requiring knowledge of reasons for tripping the Turbine / Reactor in the event of a grid disturbance.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References
EAP-APG AP/34

Student References Provided
Generator Capability Curve

APE077 AK3.01 - Generator Voltage and Electric Grid Disturbances

Knowledge of the reasons for the following responses as they apply to Generator Voltage and Electric Grid Disturbances: (CFR: 41.4, 41.5, 41.7, 41.10 / 45.8)

Reactor and turbine trip criteria…………......................................

401-9 Comments:

Remarks/Status

APE001 AA1.02 - Continuous Rod Withdrawal

Ability to operate and / or monitor the following as they apply to the Continuous Rod Withdrawal : (CFR 41.7 / 45.5 / 45.6)

Rod in-out-hold switch ....

Given the following Unit 1 conditions:

- Reactor power = 90%
- Controlling Tave fails low
- Plant Transient Response is performed
- Appropriate ICS stations are placed in MANUAL

1) Control rods are inserted to ____(1)____.

2) The RO shall inform the CRS ____(2)____ control rod insertion.

Which ONE of the following completes the statements above?

A. 1. the pre-transient rod height
   2. ONLY for the initial

B. 1. the pre-transient rod height
   2. for each

C. 1. match current feedwater demand
   2. ONLY for the initial

D. 1. match current feedwater demand
   2. for each

ILT 47 ONS SRO NRC Examination QUESTION 19

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C

General Discussion

Answer A Discussion

1st part is incorrect because per OMP 1-18, Attachment J (Plant Transient Response), Control rod are to be inserted to match feedwater demand. It is plausible because the intent of Plant Transient Response is to stabilize the plant. If feedwater was not reduced by the same instrument failure, it could be correct. Also correct under the assumption that stabilizing the plant following the failure requires returning to the pre-transient power level.

2nd part is correct per OMP 1-18.

Answer B Discussion

1st part is incorrect because per OMP 1-18, Attachment J (Plant Transient Response), Control rod are to be inserted to match feedwater demand. It is plausible because the intent of Plant Transient Response is to stabilize the plant. If feedwater was not reduced by the same instrument failure, it could be correct. Also correct under the assumption that stabilizing the plant following the failure requires returning to the pre-transient power level.

2nd part is incorrect because only the first control rod insertion for PTR si required. It is plausible because for normal evolutions, reactivity management guidelines dictate that the CRS is informed prior to reactivity insertions.

Answer C Discussion

1st part is correct. per OMP 1-18, Attachment J (Plant Transient Response), Control rod are to be inserted to match feedwater demand.

2nd part is correct per OMP 1-18.

Answer D Discussion

1st part is correct. per OMP 1-18, Attachment J (Plant Transient Response), Control rod are to be inserted to match feedwater demand.

2nd part is incorrect because only the first control rod insertion for PTR si required. It is plausible because for normal evolutions, reactivity management guidelines dictate that the CRS is informed prior to reactivity insertions.

Basis for meeting the KA

The question matches the KA by requiring knowledge of the expectations for control rod insertion during a continuous rod withdrawal event (Tave failed low).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References
OMP 1-18 Att. J ADM-OMP Pg 11

Student References Provided

APE001 AA1.02 - Continuous Rod Withdrawal

Ability to operate and / or monitor the following as they apply to the Continuous Rod Withdrawal : (CFR 41.7 / 45.5 / 45.6)

Rod in-out-hold switch ....

401-9 Comments:

Remarks/Status

ILT 47 ONS SRO NRC Examination QUESTION 20 20 **A**

APE036 AK3.02 - Fuel Handling Incidents

Knowledge of the reasons for the following responses as they apply to the Fuel Handling Incidents: (CFR 41.5,41.10 / 45.6 / 45.13)
Interlocks associated with fuel handling equipment ....

Given the following Unit 3 conditions:

- Reactor in MODE 6
- Core offload in progress
- Main Fuel Bridge is withdrawing a fuel assembly that appears to be binding

The __(1)__ interlock will stop the withdrawal of the fuel assembly to prevent fuel Damage. The load setpoint for this interlock is __(2)__.

Which ONE of the following completes the statement above?

A. 1. TS-1 (Overload Bypass)
2. 2500 lb

B. 1. TS-1 (Overload Bypass)
2. 3000 lb

C. 1. TS-2 (Hoist Interlock Bypass)
2. 2500 lb

D. 1. TS-2 (Hoist Interlock Bypass)
2. 3000 lb

General Discussion

Answer A Discussion

This is correct. 2500 lb is the interlock setpoint for the Overload TS-1.

Answer B Discussion

1st part is the correct interlock but the 2nd part is incorrect because the interlock setpoint is 2500 lb. It is plausible because this is the limit per SLC 16.12.15.

Answer C Discussion

1st part is incorrect but plausible because this interlock will also stop upward travel of the hoist however it is only in effect when the grapple is disengaged.

2nd part is correct. This is the interlock setpoint for the Overload TS-1.

Answer D Discussion

1st part is incorrect but plausible because this interlock will also stop upward travel of the hoist however it is only in effect when the grapple is disengaged.

2nd part is incorrect because the interlock setpoint is 2500 lb. It is plausible because this is the limit per SLC 16.12.15.

Basis for meeting the KA

This question matches the KA by requiring knowledge of interlocks associated with fuel handlling equipment.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT43 (Q 22)

Development References

ILT43 Q22
FH FHS Pg 34
SLC 16.12.5

Student References Provided

APE036 AK3.02 - Fuel Handling Incidents

Knowledge of the reasons for the following responses as they apply to the Fuel Handling Incidents: (CFR 41.5,41.10 / 45.6 / 45.13)

Interlocks associated with fuel handling equipment ....

401-9 Comments:

Remarks/Status

APE067 AK1.02 - Plant Fire On Site

Knowledge of the operational implications of the following concepts as they apply to Plant Fire on Site: (CFR 41.8 / 41.10 / 45.3)

Fire fighting ....

Given the following Plant conditions:

- All Units Reactor power = 100%
- 1SA3/B-6 (Fire Alarm) actuated
- AO reports flames and heavy smoke spreading to equipment and cable trays
- Fire location = Near the LPSW pumps, Column G30

Which ONE of the following locations is the affected SSF Risk Area(s) and the required action in accordance with AP/25 (Standby Shutdown Facility Emergency Operating Procedure)?

REFERENCE PROVIDED

A. ALL Three Units are affected therefore trip ALL Three Units

B. ONLY Unit 2 is affected therefore trip Unit 2 ONLY

C. ALL Three Units are affected therefore perform a controlled shutdown on all three units

D. ONLY Unit 2 is affected therefore perform a controlled shutdown on Unit 2 ONLY

General Discussion

Answer A Discussion

Incorrect: First part is incorrect. Plausible in that the fire is located partly on Unit 1 and 2 but the Attachment 1-fire plan dictates the fire as Unit 2 SSF Risk Area ONLY. The candidates must choose the correct attachment for the correct floor elevation and then determine the correct area affected. Second part is incorrect. Candidate must know manual trip criteria from AP/25 and trip only the affected unit (Unit 2 only).

Answer B Discussion

CORRECT: Per ARG for 1SA3/B-6 the fire is a Challenging Active Fire; its location is between Unit 1 and Unit 2; Attachment 1 (provided) dictates the fire as Unit 2 SSF Risk Area ONLY. AP/25 requires only Unit 2 to be manually tripped.

Answer C Discussion

Incorrect: First part is incorrect. Plausible in that the fire is located partly on Unit 1 and 2 but the Attachment 1-fire plan dictates the fire as Unit 2 SSF Risk Area ONLY. The candidates must choose the correct attachment for the correct floor elevation and then determine the correct area affected. Second part is incorrect. Plausible if the candidate does not understand there is a challenging active fire requiring a unit trip or that AP/25 requires a unit trip (since the unit trip would be a memory item). The candidate may believe a controlled shutdown is required on all three units due to LPSW being affected by the fire.

Answer D Discussion

Incorrect: First part is correct. Per ARG for 1SA3/B-6, the fire is a Challenging Active Fire. Its location is between Unit 1 and Unit 2. Attachment 1 (provided) dictates the fire as Unit 2 SSF Risk Area ONLY. Second part is incorrect. Plausible if the candidate does not understand there is a challenging active fire requiring a unit trip or that AP/25 requires a unit trip (since the unit trip would be a memory item). The candidate may believe a controlled shutdown is required on unit 2 only due to LPSW being affected by the fire.

Basis for meeting the KA

Requires knowledge of the operational implications of a fire in the plant.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	2009 (Q 73) NRC Exam

Development References

1SA3/B-6
AP 43
EAP APG
AP/25
2009 Q73

Student References Provided

AP 43 Encl 5.1

APE067 AK1.02 - Plant Fire On Site

Knowledge of the operational implications of the following concepts as they apply to Plant Fire on Site: (CFR 41.8 / 41.10 / 45.3)

Fire fighting ....

401-9 Comments:

Remarks/Status

BWA06 AK1.3 - Shutdown Outside Control Room
Knowledge of the operational implications of the following concepts as they apply to the (Shutdown Outside Control Room):
(CFR: 41.8 / 41.10 / 45.3)
Annunciators and conditions indicating signals, and remedial actions associated with the (Shutdown Outside Control Room)

Given the following Unit 1 conditions:

- The reactor was tripped
- Unit 1 & 2 Control Room was evacuated prior to any additional actions being taken
- The crew has proceeded to the Auxiliary Shutdown Panel (ASP)
- AP/8 (Loss Of Control Room) is in progress
- LDST level at the ASP = 47 inches decreasing

In accordance with AP/8:

1) LDST level will be maintained by aligning HPIP suction to the BWST____(1)____

2) If no action is taken, 1HP-24 and 1HP-25 will automatically open when LDST level decreases to a setpoint value of ____(2)____ inches.

Which ONE of the following completes the statement above?

A. 1. at the ASP
2. 40

B. 1. at the ASP
2. 45

C. 1. locally at the valve(s)
2. 40

D. 1. locally at the valve(s)
2. 45

General Discussion

Answer A Discussion

1st part is incorrect because 1HP-24 & 1HP-25 do not have controls at the ASP. It is plausible because you are directed to cycle them while shutting down from the ASP. Various other components that you are directed to manipulate at the ASP do have controls at the ASP.

2nd part is correct. In order to preserve the HPIPs, the suction sources will swap when the LDST level reaches 40 inches.

Answer B Discussion

1st part is incorrect because 1HP-24 & 1HP-25 do not have controls at the ASP. It is plausible because you are directed to cycle them while shutting down from the ASP. Various other components that you are directed to manipulate at the ASP do have controls at the ASP.

2nd part is incorrect because the valves will not open until LDST level decreases to 40 inches. It is plausible because 45 inches is the bottom of the level control band that you are directed to keep per AP/08.

Answer C Discussion

1st part is correct. 1HP-24 & 1HP-25 do not have controls at the ASP

2nd part is correct. In order to preserve the HPIPs, the suction sources will swap when the LDST level reaches 40 inches.

Answer D Discussion

1st part is correct. 1HP-24 & 1HP-25 do not have controls at the ASP

2nd part is incorrect because the valves will not open until LDST level decreases to 40 inches. It is plausible because 45 inches is the bottom of the level control band that you are directed to keep per AP/08.

Basis for meeting the KA

This question matches the KA by requiring knowledge of remedial actions associated with plant conditions when manning the Aux Shutdown Panel.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References

AP/8 Control Room Evac
EAP-APG Obj: R9
HPI Visio DWG

Student References Provided

BWA06 AK1.3 - Shutdown Outside Control Room

Knowledge of the operational implications of the following concepts as they apply to the (Shutdown Outside Control Room):
(CFR: 41.8 / 41.10 / 45.3)

Annunciators and conditions indicating signals, and remedial actions associated with the (Shutdown Outside Control Room)

401-9 Comments:

Remarks/Status

ILT 47 ONS SRO NRC Examination QUESTION 23 23 **D**

EPE074 EA1.02 - Inadequate Core Cooling

Ability to operate and monitor the following as they apply to a Inadequate Core Cooling: (CFR 41.7 / 45.5 / 45.6)

RCS cooldown rate ....

Given the following Unit 1 conditions:

Time = 0800:

- Reactor power = 100%
- Auxiliary Steam header being supplied by Unit 2
- Large Break LOCA occurs

Time = 0804:

- Transition to the ICC tab is made
- The step to reduce SG pressure is initiated

Which ONE of the following describes the guidance provided by the ICC tab?

A. SGs depressurization will be limited to 100 ^{O}F / hr cooldown rate and will stop at 250 psig.

B. SGs depressurization will be limited to 100 ^{O}F / hr cooldown rate and will continue until SGs are completely depressurized.

C. SGs depressurization will occur as rapidly as possible and will stop at 250 psig.

D. SGs depressurization will occur as rapidly as possible and will continue until SGs are completely depressurized.

ILT 47 ONS SRO NRC Examination QUESTION 23

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D

General Discussion

Answer A Discussion

1st part is incorrect because the ICC tab directs the SGs to be depressurized as rapidly as possible. It is plausible because cooldown rate is typically limited in the EOP to 100 OF/hr or less.

2nd part is incorrect because with MD EFDWPs available, the ICC tab directs the SGs to be completely depressurized. It is plausible because If the TD EFDW pump were the only feedwater pump available, it would be correct. It would also seem plausible to maintain steam pressure ~ 250 psig in case the MD EFDWPs were to fail.

Answer B Discussion

1st part is incorrect because the ICC tab directs the SGs to be depressurized as rapidly as possible. It is plausible because cooldown rate is typically limited in the EOP to 100 OF/hr or less.

2nd part is correct. If the TD EFDWP is NOT the only operating EFDWP (as in this case), then the SGs are completely depressurized.

Answer C Discussion

1st part is correct. ICC tab note prior to step 28 states that cooldown rates do NOT apply when reducing SG pressure in the following steps.

2nd part is incorrect because with MD EFDWPs available, the ICC tab directs the SGs to be completely depressurized. It is plausible because If the TD EFDW pump were the only feedwater pump available, it would be correct. It would also seem plausible to maintain steam pressure ~ 250 psig in case the MD EFDWPs were to fail.

Answer D Discussion

1st part is correct. ICC tab note prior to step 28 states that cooldown rates do NOT apply when reducing SG pressure in the following steps.

2nd part is correct. If the TD EFDWP is NOT the only operating EFDWP (as in this case), then the SGs are completely depressurized.

Basis for meeting the KA

This question matches the KA by requiring knowledge of RCS cooldown rates when in the Inadequate Core Cooling tab of the EOP.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References
EAP ICC Pg 15 EOP ICC Tab

Student References Provided

EPE074 EA1.02 - Inadequate Core Cooling

Ability to operate and monitor the following as they apply to a Inadequate Core Cooling: (CFR 41.7 / 45.5 / 45.6)

RCS cooldown rate ....

401-9 Comments:	Remarks/Status

BWA01 AK3.3 - Plant Runback

Knowledge of the reasons for the following responses as they apply to the (Plant Runback)

(CFR: 41.5 / 41.10, 45.6, 45.13)

Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.

Given the following Unit 1 conditions:

- Reactor power = 80% stable
- ΔTc Controller is in HAND
- 1B1 RCP trips
- Crew performs Plant Transient Response
- Crew enters AP/1 (Unit Runback)
- ΔTc = +1.2 and becoming more positive

The operator will have to manually re-ratio feedwater such that feed to the __(1)__ SG will increase because the ___(2)___.

Which ONE of the following completes the above statement?

A. 1. 1A
2. RC Flow Ratio circuit has failed

B. 1. 1A
2. RC Flow Ratio circuit is blocked when the Delta Tc controller is in HAND

C. 1. 1B
2. RC Flow Ratio circuit has failed

D. 1. 1B
2. RC Flow Ratio circuit is blocked when the Delta Tc controller is in HAND

General Discussion

Answer A Discussion

First part is correct. DTc = Tc (a loop) - Tc(b loop). If the re-ratio does not occur when the 1B1 RCP trips, the B SG has too muchfeed/steam flow hich cools off the B loop. This will be seen as a + DTc.

2nd part is correct. The RC Flow Ratio circuit taps in down stream of the DTc controller so the controller being in HAND will not stop the RC Flow Ratio circuit from providing input to the summer. If this does not happen when RC flow changes, the RC Flow Ratio circuit has failed.

Answer B Discussion

First part is correct. DTc = Tc (a loop) - Tc(b loop). If the re-ratio does not occur when the 1B1 RCP trips, the B SG has too muchfeed/steam flow hich cools off the B loop. This will be seen as a + DTc.

2nd part is incorrect because its output taps in down stream of the DTc controller. It is plausible because it, as well as the DTc controller provides input to the re-ratio summer. It would be logical to think that its input woul dbe controlled by the DTc controller.

Answer C Discussion

1st part is incorrect. DTc = Tc (a loop) - Tc(b loop). If the re-ratio does not occur when the 1B1 RCP trips, the B SG has too muchfeed/steam flow hich cools off the B loop. This will be seen as a + DTc. It is plausible because it a common misconception to think that a + DTc means that the A SG needs less feedwater.

2nd part is correct. The RC Flow Ratio circuit taps in down stream of the DTc controller so the controller being in HAND will not stop the RC Flow Ratio circuit from providing input to the summer. If this does not happen when RC flow changes, the RC Flow Ratio circuit has failed.

Answer D Discussion

1st part is incorrect. DTc = Tc (a loop) - Tc(b loop). If the re-ratio does not occur when the 1B1 RCP trips, the B SG has too muchfeed/steam flow hich cools off the B loop. This will be seen as a + DTc. It is plausible because it a common misconception to think that a + DTc means that the A SG needs less feedwater.

2nd part is incorrect because its output taps in down stream of the DTc controller. It is plausible because it, as well as the DTc controller provides input to the re-ratio summer. It would be logical to think that its input woul dbe controlled by the DTc controller..

Basis for meeting the KA

Question requires evaluating the plant response and determining the reason for the plant behavior.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References

EAP-APG
AP/1
ICS Ch 4
ICS Feedwater Visio
SAEL 132 Sim Trn
ICS Feedwater Power Point

Student References Provided

BWA01 AK3.3 - Plant Runback

Knowledge of the reasons for the following responses as they apply to the (Plant Runback)

ILT 47 ONS SRO NRC Examination QUESTION 24 24 **A**

(CFR: 41.5 / 41.10, 45.6, 45.13)

Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.

401-9 Comments:	Remarks/Status

BWA04 AK2.2 - Turbine Trip

Knowledge of the interrelations between the (Turbine Trip) and the following:
(CFR: 41.7 / 45.7)

Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Given the following Unit 1 conditions:
Time = 1200

- Reactor power = 40%
- PCB 20 and PCB 21, Generator Output Breakers open
- A plant runback initiates

Time = 1202

- Reactor power = 30% decreasing
- Main Turbine trips

At Time = 1204, the SGs will be fed from ____(1)____ feedwater and heat removal from the reactor will be by ____(2)____circulation.

Which ONE of the following completest the statements above?

A. 1. Main
2. Natural

B. 1. Main
2. Forced

C. 1. Emergency
2. Natural

D. 1. Emergency
2. Forced

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D

General Discussion

Answer A Discussion

1st part is incorrect because the condensate pumps will trip when the turbine trips. The FDW pumps will trip ~ 90 seconds later. EFDW pumps will then start and feed the SGs. It is plausible because at any time < 90 seconds after the trip, it could be correct.

2nd part is incorrect because the RCPs will still be running. It is plausible because a slow transfer occurs which means that the RCP busses 1TA & 1TB will be without power for ~ 1.7 seconds. The breakers will remain closed however and the RCPs will remain operating when power returns.

Answer B Discussion

1st part is incorrect because the condensate pumps will trip when the turbine trips. The FDW pumps will trip ~ 90 seconds later. EFDW pumps will then start and feed the SGs. It is plausible because at any time < 90 seconds after the trip, it could be correct.

2nd part is correct. The RCPs will stil be running.

Answer C Discussion

1st part is correct. The condensate pumps will trip when the turbine trips. The FDW pumps will trip ~ 90 seconds later. EFDW pumps will then start and feed the SGs.

2nd part is incorrect because the RCPs will still be running. It is plausible because a slow transfer occurs which means that the RCP busses 1TA & 1TB will be without power for ~ 1.7 seconds. The breakers will remain closed however and the RCPs will remain operating when power returns.

Answer D Discussion

1st part is correct. The condensate pumps will trip when the turbine trips. The FDW pumps will trip ~ 90 seconds later. EFDW pumps will then start and feed the SGs.

2nd part is correct. The RCPs will stil be running.

Basis for meeting the KA

This question matches the KA by requiring knowledge of how heat is removed form the plant in the event of a turbine trip.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References
EL EPD Pg 104, 105 AP 1

Student References Provided

BWA04 AK2.2 - Turbine Trip

Knowledge of the interrelations between the (Turbine Trip) and the following:
(CFR: 41.7 / 45.7)

Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

401-9 Comments:

Remarks/Status

D

BWE08 EA2.2 - LOCA Cooldown
Ability to determine and interpret the following as they apply to the (LOCA Cooldown)
(CFR: 43.5 / 45.13)
Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Given the following Unit 1 conditions:

- A SB LOCA has occurred
- LOCA CD tab in progress
- 1A LPI Pump operating in the Piggyback alignment

Which ONE of the following describes the:

1) operational limitations on the operating LPI pump?

2) pump(s) being protected by the above limitation?

A. 1. Maximized to < 3100 gpm
2. LPI

B. 1. Maximized to < 3100 gpm
2. HPI

C. 1. Maximized to < 2900 gpm
2. LPI

D. 1. Maximized to < 2900 gpm
2. HPI

General Discussion

Answer A Discussion

Correct. With only one LPI pump operating in the Piggyback mode LPI flow is maximized to < 3100 gpm to protect the LPI pump from runout.

Answer B Discussion

First part is correct. Limit for 1 LPI pump is 3100 gpm.

Second part is incorrect because the limit is for the LPI pump. It is plausible since the LPI pump is supplying suction to the HPI pumps in this alignment and other conditions place strict flow limits on the HPI pumps to protect them from damage.

Answer C Discussion

First part is incorrect because the limit is 3100 gpm for the LPI pump. It is plausible since 2900 gpm is a flow limit applicable when only one LPI train is operating however it is the LPI flow that transitions the mitigation strategy to a LBLOCA from a SBLOCA or allows securing HPI pumps following a SBLOCA..

Second part is correct. This is a flow limit for the LPI pump.

Answer D Discussion

First part is incorrect because the limit is 3100 gpm for the LPI pump. It is plausible since 2900 gpm is a flow limit applicable when only one LPI train is operating however it is the LPI flow that transitions the mitigation strategy to a LBLOCA from a SBLOCA or allows securing HPI pumps following a SBLOCA..

Second part is incorrect because the limit is for the LPI pump. It is plausible since the LPI pump is supplying suction to the HPI pumps in this alignment and other conditions place strict flow limits on the HPI pumps to protect them from damage.

Basis for meeting the KA

Matches the KA by requiring knowledge of limitations on plant components when performing plant procedures.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT44 (Q 26) NRC Exam

Development References

ILT44 Q26
LOCA CD Tab
EAP LOCA CD Pg 10

Student References Provided

BWE08 EA2.2 - LOCA Cooldown
Ability to determine and interpret the following as they apply to the (LOCA Cooldown)
(CFR: 43.5 / 45.13)
Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

401-9 Comments:

Remarks/Status

ILT 47 ONS SRO NRC Examination QUESTION 27

27

B

BWE14 2.4.50 - EOP Enclosures

BWE14 GENERIC

Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 41.10 / 43.5 / 45.3)

1SA-1

	1	2	3	4	5	6	7	8	9	10	11	12
A	1 A RPS TRIP	1 A LO PRESS TRIP	1A FLUX/IMB/FLOW TRIP	1A HI TEMP TRIP	1A VAR LO PRESS TRIP	1A HI PRESS TRIP	1A RCP/ FLUX TRIP	1A HI FLUX TRIP	1A R.B. HI PRESS TRIP	ES 1 TRIP	ES 5 TRIP	ICS LOSS OF ACS POWER FUSE BLOWN
B	1 B RPS TRIP	1B LO PRESS TRIP	1B FLUX/IMB/FLOW TRIP	1B HI TEMP TRIP	1B VAR LO PRESS TRIP	1B HI PRESS TRIP	1B RCP/ FLUX TRIP	1B HI FLUX TRIP	1B R.B. HI PRESS TRIP	ES 2 TRIP	ES 6 TRIP	ICS AUTO/HAND POWER FUSE BLOWN
C	1 C RPS TRIP	1 C LO PRESS TRIP	1C FLUX/IMB/FLOW TRIP	1C HI TEMP TRIP	1C VAR LO PRESS TRIP	1C HI PRESS TRIP	1C RCP/ FLUX TRIP	1C HI FLUX TRIP	1C R.B. HI PRESS TRIP	ES 3 TRIP	ES 7 TRIP	LP INJECTION PUMP "A" DIFF. PRESS LOW
D	1 DRPS TRIP	1 D LO PRESS TRIP	1D FLUX/IMB/FLOW TRIP	1D HI TEMP TRIP	1A VAR LO PRESS TRIP	1D HI PRESS TRIP	1D RCP/ FLUX TRIP	1D HI FLUX TRIP	1D R.B. HI PRESS TRIP	ES 4 TRIP	ES 8 TRIP	LP INJECTION PUMP "B" DIFF. PRESS LOW
E	CRD SEQUENCE FAULT	CRD TRIP BKR A TRIP	CRD TRIP BKR B TRIP	CRD TRIP BKR C TRIP	CRD TRIP BKR D TRIP	CRD ELECTRONIC TRIP E	CRD ELECTRONIC TRIP F	DIVERSE HPI BYP	DIVERSE HPI TRIP	DIVERSE LPI BYP	DIVERSE LPI TRIP	LP INJECTION PUMP "C" DIFF. PRESS LOW

Given the following Unit 1 conditions

Initial conditions:

- Reactor power = 45% stable

Current conditions:

- Reactor power = <1% WR decreasing
- Core SCM = 0°F stable
- RCS pressure = 140 psig decreasing
- Reactor Building pressure = 16.4 psig increasing
- 1SA-1 alarms as indicated above

Which ONE of the following describes actions required by the EOP?

A. Secure running LPI pumps

B. Manually actuate ES Digital Channels 7 & 8

C. Dispatch AO to open CRD breakers C & D

D. Feed to LOSCM setpoint with Emergency Feedwater

General Discussion

Answer A Discussion

Incorrect. Plausible since it would be correct if RCS pressure were > 200 psig so that the pumps would be running against shutoff head.

Answer B Discussion

Correct. With RB pressure > 10 psig, ES 7 & 8 should have actuated. When performing Encl. 5.1 you will be directed to manually actuate ES 7&8 if they have not already actuated.

Answer C Discussion

Incorrect.. Plausible since it would be correct of power were still >5%. Additionally plausible since only two of the four CRD breakers indicate tripped.

Answer D Discussion

Incorrect. Plausible since it would be correct if RCS pressure were above 200 psig (and therefore no LPI flow). In this case, RCS pressure is low enough to have sufficient LPI flow which means that Rule 2 and the LOSCM tab will not direct feeding to LOSCM stpt.

Basis for meeting the KA

Requires verifying alarms actuated on 1SA-1 are appropriate for plant conditions. ES-7&8 should be actuated with RB pressure > 10 psig. Alarm response states ES equipment that should have operated and per OMP1-18, equipment that should have operated and did not (RPS/ES), it should be manually initated.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	2009B (Q 41) NRC Exam

Development References

IC-ES Pg 15
EOP Encl. 5.1
2009B Q41

Student References Provided

BWE14 2.4.50 - EOP Enclosures
BWE14 GENERIC
Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 41.10 / 43.5 / 45.3)

401-9 Comments:

Remarks/Status

SYS064 K1.02 - Emergency Diesel Generator (ED/G) System

Knowledge of the physical connections and/or cause-effect relationships between the ED/G system and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

D/G cooling water system ....

Plant conditions:

- It is desired to perform a manual start of KHU-1 from the Unit 1/2 Control Room
- The MASTER TRANSFER switch for KHU-1 is positioned to REMOTE
- UNIT 1 MASTER SELECTOR is placed in MAN
- UNIT 1 SYNC 230 KV selector is placed in MAN
- UNIT 1 LOCAL MASTER switch is placed in START and held in position for > 10 seconds until KHU-1 starts

In the above starting sequence, which ONE of the following is correct regarding generator cooling water flow?

A. When the MASTER SELECTOR switch is placed in MAN, the generator cooling water pump will start to provide water to the cooler.

B. When the MASTER SELECTOR switch is placed in MAN, the generator cooling water valve will open to provide water to the cooler.

C. When the LOCAL MASTER switch is placed in START, the generator cooling water pump will start to provide water to the cooler.

D. When the LOCAL MASTER switch is placed in START, the generator cooling water valve will open to provide water to the cooler.

General Discussion

Answer A Discussion

Incorrect because the cooling water flow is via gravity feed. When the Master Selector is placed in MAN, the valve opens to provide cooling water flow to the ventilation system and the thrust bearing oil cooler.

Answer B Discussion

Correct. When the Master Selector is placed in MAN, the valve opens to provide cooling water flow to the ventilation system and the thrust bearing oil cooler.

Answer C Discussion

Incorrect because cooling water flow will be supplied when the MASTER SELECTOR is placed in MAN. It is plausible because typically cooling water is supplied on generator start (like SG service water) which will use a pump instead of gravity feed.

Answer D Discussion

Incorrect because cooling water flow will be supplied when the MASTER SELECTOR is placed in MAN. It is plausible because typically cooling water is supplied on generator start (like SG service water).

Basis for meeting the KA

This question matches the KA by requiring knowledge of the relationship between the KHU and cooling water during a generator start.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	2010A (Q 29) NRC Exam

Development References

EL KHG Pg 17
KHU Start procedure

Student References Provided

SYS064 K1.02 - Emergency Diesel Generator (ED/G) System

Knowledge of the physical connections and/or cause-effect relationships between the ED/G system and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

D/G cooling water system ....

401-9 Comments:

Remarks/Status

SYS003 K6.04 - Reactor Coolant Pump System (RCPS)
Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: (CFR: 41.7 / 45/5)
Containment isolation valves affecting RCP operation ....

Given the following Unit 1 conditions:
Time = 1200

- Reactor power = 65% stable
- 1LPSW-6 (UNIT 1 RCP COOLERS SUPPLY) fails closed

Time = 1205

- AP/16 (Abnormal RCP Operation) in progress
- RCP Temperatures:

	1A1	**1A2**	**1B1**	**1B2**
Upper Guide Bearing Temp	182ºF	197ºF	188ºF	185ºF
Seal Return Temp	169ºF	174ºF	227ºF	187ºF

Which ONE of the following is required per AP/16 at Time = 1205?

A. Manually trip the Reactor and stop ALL RCPs

B. Manually trip the Reactor and stop RCPs 1A2 & 1B1 ONLY

C. Stop RCP 1A2 ONLY and verify FDW re-ratios properly

D. Stop RCP 1B1 ONLY and verify FDW re-ratios properly

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General Discussion

Answer A Discussion

Incorrect because a reactor trip is not required and all RCPs are not required to be tripped. It is plausible because with LPSW-6 failing closed, its cooling has been lost to all RCPs. There is no provision in AP/16 to trip RCPs based on a loss of LPSW cooling.

Answer B Discussion

Incorrect because RCP 1B1 is not required to be tripped. While its Seal Return Temperature is high, it is still below the trip criteria. If it were met, this answer would be correct.

Answer C Discussion

Correct. RCP 1A2 is the only pump that exceeds the RCP immediate trip criteria (Upper Guide Bearing temp > 190 degrees). All other parameters provided are below the immediate trip criteria. Since 3 RCPs will still be operating, a reactor trip is not required.

Answer D Discussion

Incorrect because 1B1 RCP parameters are below RCP immediate trip criteria. It is plausible because 227 degrees is above the trip criteria for several RCP parameters but not Seal Return temperaure.

Basis for meeting the KA

Requires the ability to monitor RCP motor parameters and determine that two pumps exceed temperature limits of AP/16. The limits of AP/16 also must be know by the student.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT39 (Q 28) NRC Exam

Development References

AP/16
EAP-APG R9
ILT39 Q28

Student References Provided

SYS003 K6.04 - Reactor Coolant Pump System (RCPS)

Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: (CFR: 41.7 / 45/5)

Containment isolation valves affecting RCP operation ....

401-9 Comments:

Remarks/Status

ILT 47 ONS SRO NRC Examination QUESTION 30 30 **C**

SYS004 A2.15 - Chemical and Volume Control System

Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5/ 43/5 / 45/3 / 45/5)

High or low PZR level ....

Given the following Unit 1 conditions:
Initial conditions:

- Reactor power = 100%
- LDST level = 90" Stable

Current conditions:

- 1HP-14 fails to the Bleed position

1) Over the next 5 minutes, 1HP-120 will __ (1) __ to maintain Pzr level constant.

2) __ (2) __ will be entered to mitigate this event.

Which ONE of the following completes the statements above?

A. 1. open
2. AP/02 (Excessive Leakage)

B. 1. open
2. AP/32 (Loss of Letdown)

C. 1. remain in its current position
2. AP/02 (Excessive Leakage)

D. 1. remain in its current position
2. AP/32 (Loss of Letdown)

General Discussion

Answer A Discussion

Incorrect. First part is plausible because a leak from the RCS exists. However since Letdown flow is being diverted LDST level will decrease but Pzr level will not be affected.

Second part is correct. This meets entry conditions for AP/2.

Answer B Discussion

First part is incorrect because 1HP-120 will remain in its current position. It is plausible because a leak from the RCS exists. However since Letdown flow is being diverted LDST level will decrease but Pzr level will not be affected.

Second part is incorrect because AP/2 will be entered. It is plausible because Letdown flow is affected but is not lost.

Answer C Discussion

Correct. When 1HP-14 fails in the Bleed position letdown flow will be diverted to the BHUT. This will cause LDST level to decrease but Pzr level will not be affected. !HP-14 failing to the Bleed position is an entry conditions of AP/002.

Answer D Discussion

1st part is correct. When 1HP-14 fails in the Bleed position letdown flow will be diverted to the BHUT. This will cause LDST level to decrease but Pzr level will not be affected.

Second part is plausible because Letdown flow is affected but is not lost.

Basis for meeting the KA

Question requires knowledge of LCS response to a failure of 1HP-14. 1HP-14 is a part of the CVCS.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT41 (Q 29) NRC Exam

Development References

PNS-HPI Pg 21
AP 2 Entry Conditions
ILT41 Q29

Student References Provided

SYS004 A2.15 - Chemical and Volume Control System

Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5/ 43/5 / 45/3 / 45/5)

High or low PZR level ....

401-9 Comments:

Remarks/Status

SYS005 K4.08 - Residual Heat Removal System (RHRS)

Knowledge of RHRS design feature(s) and/or interlock(s) which provide or the following : (CFR: 41.7)

Lineup for "piggy-back" mode with high-pressure injection ....

Given the following Unit 1 conditions:

- Reactor power = 100%
- A LOCA occurs
- Rule 2 (Loss of SCM) is initiated
- RCS pressure = 1500 psig slowly decreasing
- 1HP-24 and 1HP-25 fail to open

When the "Piggyback" lineup is <u>complete</u>, there will be ____(1)____ LPI pump(s) and ____(2)____ HPI pumps operating.

Which ONE of the following completes the statement above?

A. 1. one
   2. two

B. 1. one
   2. three

C. 1. two
   2. two

D. 1. two
   2. three

General Discussion

Answer A Discussion

1st part is correct. Per Rule 2, 2 LPI pumps are started initially, then if LPI is only needed for Piggyback operation, it directs securing one of the LPI pumps.

2nd part is incorrect but plausible since it would be correct if only one of the BWST suction valves had failed to open.

Answer B Discussion

1st part is correct. Per Rule 2, 2 LPI pumps are started initially, then if LPI is only needed for Piggyback operation, it directs securing one of the LPI pumps.

2nd part is correct. With both BWST suction vavles failed "all available" HPI pumps are started and left running.

Answer C Discussion

1st part is incorrect because when the lineup is complete, there is only one LPI pump operating. It is plausible because when initiating the piggyback lineup, Rule 2 directs starting 2 LPI pumps. It then states that if the LPI pumps are only operating to support the piggyback lineup, secure 1 of the LPI pumps. With RCS pressure at 1500 psig, it is not providing any makeup function.

2nd part is incorrect but plausible since it would be correct if only one of the BWST suction valves had failed to open.

Answer D Discussion

1st part is incorrect because when the lineup is complete, there is only one LPI pump operating. It is plausible because when initiating the piggyback lineup, Rule 2 directs starting 2 LPI pumps. It then states that if the LPI pumps are only operating to support the piggyback lineup, secure 1 of the LPI pumps. With RCS pressure at 1500 psig, it is not providing any makeup function.

2nd part is correct. With both BWST suction vavles failed "all available" HPI pumps are started and left running.

Basis for meeting the KA

This question matches the KA by requiring knowledge of the LPI (RHR) system design and how it is orientated for piggyback operation.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

PNS LPI Pg 42, 43
Rule 2 LOSCM
DWG LPI Piggyback

Student References Provided

SYS005 K4.08 - Residual Heat Removal System (RHRS)

Knowledge of RHRS design feature(s) and/or interlock(s) which provide or the following : (CFR: 41.7)

Lineup for "piggy-back" mode with high-pressure injection ....

401-9 Comments:

Remarks/Status

SYS006 K2.04 - Emergency Core Cooling System (ECCS)
Knowledge of bus power supplies to the following: (CFR: 41.7)
ESFAS-operated valves ....

Which ONE of the following states the power supply for 3LP-18?

A. 3XS-1

B. 3XS-2

C. 3XS-3

D. 3XS-4

General Discussion

Answer A Discussion

Incorrect: Plausible since it would be correct for 1LP-17

Answer B Discussion

Correct: 1LP-18 is powered from 3XS-2.

Answer C Discussion

Incorrect: Plausible since it would be correct for 1HP-409/410

Answer D Discussion

Incorrect: Plausible since it would be correct for 1LP-19

Basis for meeting the KA

Requires knowledge of bus power supply's for ESFAS valves.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT40 (Q 33) NRC Exam

Development References

IC-ES Obj: R20, Pg 39
ILT40 Q33

Student References Provided

SYS006 K2.04 - Emergency Core Cooling System (ECCS)
Knowledge of bus power supplies to the following: (CFR: 41.7)
ESFAS-operated valves ....

401-9 Comments:

Remarks/Status

SYS007 K3.01 - Pressurizer Relief Tank/Quench Tank System (PRTS)

Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: (CFR: 41.7 / 45.6)

Containment ....

Given the following Unit 1 conditions:
Initial conditions:

- Loss of all Feedwater
- HPI forced cooling initiated
- Quench Tank pressure = 50 psig increasing
- RCS activity indicates no fuel failures present

Current conditions:

- Quench Tank pressure = 3 psig stable

Which ONE of the following describes the containment response?

A. RB Normal sump level rises and 1RIA-47 radiation level increases

B. RB Normal sump level rises and 1RIA-47 radiation level remains constant

C. RB Normal sump level remains constant and 1RIA-47 radiation level increases

D. RB Normal sump level remains constant and 1RIA-47 radiation level remains constant

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A

General Discussion

Answer A Discussion

Correct. Decrease in Quench Tank pressure indicates the Rupture Disk has blown. Inventory from the Quench Tank will go to the RBNS causing a level increase. RCS activity in the inventory will result in 1RIA-47 reading increase.

Answer B Discussion

Incorrect. RBNS response is correct. 1RIA-47 response is incorrect but plausible if RCS activity is assumed to be negligible or the source of QT pressure rise is due to DW/B Bleed in-leakage. (OE)

Answer C Discussion

Incorrect. RBNS response is incorrect but plausible if the pressure reduction is assumed to be caused by draining to the Misc Waste System via the Component Drain flow path. 1RIA-47 response is correct.

Answer D Discussion

Incorrect. RBNS response is incorrect as noted above. 1RIA-47 response is consistent with inventory going to Misc Waste or assuming activity is negligible (OE with Demin Water & B Bleed Holdup Tank water leak into the Quench Tank).

Basis for meeting the KA

Requires knowledge of impact of discharge from PORV to the Quench Tank and indications of failed/blown rupture disk and the impact of the failure on containment parameters.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	2009 (Q 32) NRC Exam

Development References

PNS-PZR Obj: R12
Sys Dwg
2009 Q32

Student References Provided

SYS007 K3.01 - Pressurizer Relief Tank/Quench Tank System (PRTS)

Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: (CFR: 41.7 / 45.6)

Containment ....

401-9 Comments:

Remarks/Status

SYS007 K5.02 - Pressurizer Relief Tank/Quench Tank System (PRTS)
Knowledge of the operational implications of the following concepts as they apply to PRTS: (CFR: 41.5 / 45.7)
Method of forming a steam bubble in the PZR ....

Given the following Unit 1 conditions:

- OP/1/A/1103/002, (Filling and Venting RCS) Enclosure 4.14 (Establishing Pzr Steam Bubble And RCS Final Vent) in progress
- The Pressurizer is vented to the Quench Tank for 30 minutes

1) Quench Tank level should increase a minimum of ____(1)____ to indicate that Pzr Steam Bubble Formation is complete?

2) A consequence of incomplete Pzr bubble formation is that ____(2)____.

Which ONE of the following completes the statements above?

A. 1. 0.2 inches
   2. Pzr spray will NOT effectively control RCS pressure on an insurge

B. 1. 0.2 inches
   2. Pzr heaters will NOT effectively control RCS pressure on an outsurge

C. 1. 2.0 inches
   2. Pzr spray will NOT effectively control RCS pressure on an insurge

D. 1. 2.0 inches
   2. Pzr heaters will NOT effectively control RCS pressure on an outsurge

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General Discussion

Answer A Discussion

1st part is incorrect because the increase in level is ~ 2 inches. It is plausible because the pressure increase should be < 0.2 psi. The number is used in the procedure but for pressure not level.

2nd part is correct. Condensing steam is what gives the pressurizer its ability to mitigate plant transients.

Answer B Discussion

1st part is incorrect because the increase in level is ~ 2 inches. It is plausible because the pressure increase should be < 0.2 psi. The number is used in the procedure but for pressure not level.

2nd part is incorrect because Pzr heater will still heat up and re-prerssurize the Pzr/RCS during an outsurge however, the steam bubble's ability to mitigate the pressure drop is deminished.

Answer C Discussion

1st part is correct. Per OP/1/A/1103/002, a Pzr level increase of at least 2.0 inches in the QT is an indication of Pzr bubble formation.

2nd part is correct. Condensing steam is what gives the pressurizer its ability to mitigate plant transients.

Answer D Discussion

1st part is correct. Per OP/1/A/1103/002, a Pzr level increase of at least 2.0 inches in the QT is an indication of Pzr bubble formation.

2nd part is incorrect because Pzr heater will still heat up and re-prerssurize the Pzr/RCS during an outsurge however, the steam bubble's ability to mitigate the pressure drop is deminished.

Basis for meeting the KA

Requires knowledge of the QT operational parameters (pressure and level changes)
that indicate Pzr steam bubble formation is complete

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	ILT41 (Q 33) NRC Exam

Development References

OP/1/A/1103/002, Encl. 4.10
PNS-PZR Obj R17, Pg 32
ILT41 Q33

Student References Provided

SYS007 K5.02 - Pressurizer Relief Tank/Quench Tank System (PRTS)
Knowledge of the operational implications of the following concepts as they apply to PRTS: (CFR: 41.5 / 45.7)
Method of forming a steam bubble in the PZR ....

401-9 Comments:

Remarks/Status

SYS008 A1.01 - Component Cooling Water System (CCWS)

Ability to predict and/or monitor changes in parameters to prevent exceeding design limits) associated with operating the CCWS controls including : (CFR: 41.5 / 45.5)

CCW flow rate ....

Given the following Unit 1 Conditions:

- Reactor power = 100%

1) ___(1)___ would result in an increase in CC Cooler outlet temperature ºF.

2) The Component Cooling water temperature exiting the Letdown Cooler is monitored by ___(2)___.

Which ONE of the following completes the statements above?

A. 1. Throttling open 1HP-7
2. OAC indication ONLY

B. 1. Throttling open 1HP-7
2. OAC indication AND Control Room temperature gage

C. 1. Placing 1HP-14 in “BLEED”
2. OAC indication ONLY

D. 1. Placing 1HP-14 in “BLEED”
2. OAC indication AND Control Room temperature gage

General Discussion

Answer A Discussion

1st part is correct. Opening HP-7 would increase the flow of the hot fluid which would, in turn incrase the outlet temperatures of both fluids.

2nd part is correct. This parameter is not displayed on a control room gauge.

Answer B Discussion

1st part is correct. Opening HP-7 would increase the flow of the hot fluid which would, in turn incrase the outlet temperatures of both fluids.

Second part incorrect because there is no control room gauge for this parameter. It is plausible because other CC system parameters (surge tank level, CC flow) have both an OAC indication and a gage in the control room.

Answer C Discussion

1st part is incorrect because the the letdown flowrate does not change when going to bleed. It is plausible because it may seem logical that valving in a tank that may be at a lower pressure would incrase the letdown flowrate.

2nd part is correct. This parameter is not displayed on a control room gauge.

Answer D Discussion

1st part is incorrect because the the letdown flowrate does not change when going to bleed. It is plausible because it may seem logical that valving in a tank that may be at a lower pressure would incrase the letdown flowrate.

Second part incorrect because there is no control room gauge for this parameter. It is plausible because other CC system parameters (surge tank level, CC flow) have both an OAC indication and a gage in the control room.

Basis for meeting the KA

The question requires the applicant to recognize that the CC system temperature, as monitored by the OAC indication of the CC System Letdown Cooler Outlet temperature, should be kept below 225F.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT43 (Q 34) NRC Exam

Development References

OC-OP-PNS-CC Obj: R8, Pg 15, 16
CC System dwg
HPI System dwg
ILT43 Q34

Student References Provided

SYS008 A1.01 - Component Cooling Water System (CCWS)

Ability to predict and/or monitor changes in parameters to prevent exceeding design limits) associated with operating the CCWS controls including : (CFR: 41.5 / 45.5)

CCW flow rate ....

401-9 Comments:

Remarks/Status

SYS010 K5.01 - Pressurizer Pressure Control System (PZR PCS)

Knowledge of the operational implications of the following concepts as the apply to the PZR PCS: (CFR: 41.5 / 45.7)

Determination of condition of fluid in PZR, using steam tables ....

Given the following Unit 3 conditions:

Initial conditions:

- Reactor power = 100%
- Switchyard Isolation occurs

Current Conditions:

- Natural Circulation established
- RCS pressure = 2155 psig
- Tcold = 550°F stable
- Pressurizer level = 220” stable
- Pressurizer temperature = 628°F

1) The Pressurizer is __(1)__.

2) Pressurizer Heater Bank #2 (Groups B & D) heaters are __(2)__.

Which ONE of the following completes the statements above?

A. 1. saturated
   2. energized

B. 1. subcooled
   2. energized

C. 1. saturated
   2. NOT energized

D. 1. subcooled
   2. NOT energized

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General Discussion

Answer A Discussion

First part is incorrect because the Pzr is subcooled. It is plausible since it would be correct for normal Pzr temperatures. With RCS pressure, Tcold, and Pzr level at their normal values it is plausible to believe that the Pzr is in its normal state of saturated.

Second part is correct. Bank 2 heaters are used in the Pzr saturation recovery circuit. As long as RCS pressure is at least 20 psig from saturation pressure of the Pzr these heaters would be energized. Additionally, the heaters are fed from 1X8 which do not load shed therefore even following the Switchyard isolation, the heaters would be energized since the Pzr is subcolled by about 350 psig.

Answer B Discussion

1st part is correct: With RCS pressure at 2150 psig, saturation temperature for that pressure is approximately 648 degrees F. With the Pressurizer temp at 628 degrees, the Pzr is subcooled.

2nd part is correct. Bank 2 heaters are used in the Pzr saturation recovery circuit. As long as RCS pressure is at least 20 psig from saturation pressure of the Pzr these heaters would be energized. Additionally, the heaters are fed from 1X8 which do not load shed therefore even following the Switchyard isolation, the heaters would be energized since the Pzr is
subcolled by about 350 psig.

Answer C Discussion

First part is incorrect because the Pzr is subcooled. It is plausible since it would be correct for normal Pzr temperatures. With RCS pressure, Tcold, and Pzr level at their normal values it is plausible to believe that the Pzr is in its normal state of saturated.

Second part is incorrect because the heater group B & D would be energized. It is plausible since RCS pressure is at 2155 therefore is the Pzr were actually saturated the Bank 2 heaters would be OFF since they turn off at 2150 psig.

Answer D Discussion

1st part is correct: With RCS pressure at 2150 psig, saturation temperature for that pressure is approximately 648 degrees F. With the Pressurizer temp at 628 degrees, the Pzr is subcooled.

Second part is incorrect because the heater group B & D would be energized. It is plausible since RCS pressure is at 2155 therefore is the Pzr were actually saturated the Bank 2 heaters would be OFF since they turn off at 2150 psig.

Basis for meeting the KA

This question requires determining that the Pzr is Subcooled using steam tables and the status of Pzr heaters.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT40 (Q 36) NRC Exam

Development References
PNS-Pzr Obj: R5, Pg 19, 40 Steam Tables ILT40 Q36

Student References Provided
Steam Tables

SYS010 K5.01 - Pressurizer Pressure Control System (PZR PCS)

Knowledge of the operational implications of the following concepts as the apply to the PZR PCS: (CFR: 41.5 / 45.7)

Determination of condition of fluid in PZR, using steam tables ....

401-9 Comments:	Remarks/Status

SYS010 A3.01 - Pressurizer Pressure Control System (PZR PCS)
Ability to monitor automatic operation of the PZR PCS, including: (CFR: 41.7 / 45.5)
PRT temperature and pressure during PORV testing ....

Given the following Unit 1 conditions:

- Mode 5
- Pzr bubble has just been established
- PT/1/A/020/0201/004 (RC-66 Stroke Test) is being performed
- Pzr pressure = 40 psig
- Quench Tank pressure = 0 psig

When RC-66 is opened QT pressure will ____(1)____ and the temperature downstream of the PORV will increase to ____(2)____.

Which ONE of the following completes the statement above?

A. 1. remain approximately the same
2. 212 ^{O}F

B. 1. remain approximately the same
2. 260 ^{O}F

C. 1. increase
2. 212 ^{O}F

D. 1. increase
2. 260 ^{O}F

General Discussion

Answer A Discussion

1st part is correct. With steam in the Pzr, it should condense when discharging into the QT and for the time that the PORV is open, there should not be a pressure increase.

2nd part is incorrect because temperature downstream of the PORV should be ~ 260 degrees. It is plausible because if Pzr pressure were > ~ 1800 psig, it would be correct. The majority of these types of problems start with the PORV at NOP and NOT so the discharge will always be at sat temperature for the QT (212 degrees if > 1800 psig).

Answer B Discussion

1st part is correct. With steam in the Pzr, it should condense when discharging into the QT and for the time that the PORV is open, there should not be a pressure increase.

2nd part is correct. This is the calculated temperature for the stated conditions.

Answer C Discussion

1st part is incorrect because you would not expect QT pressure to increase. It is plausible because if there were still N2 in the Pzr, it would be correct. A prolonged period of time with the PORV open would cause QT pressure to increase as the water temperature increased.

2nd part is incorrect because temperature downstream of the PORV should be ~ 260 degrees. It is plausible because if Pzr pressure were > ~ 1800 psig, it would be correct. The majority of these types of problems start with the PORV at NOP and NOT so the discharge will always be at sat temperature for the QT (212 degrees if > 1800 psig).

Answer D Discussion

1st part is incorrect because you would not expect QT pressure to increase. It is plausible because if there were still N2 in the Pzr, it would be correct. A prolonged period of time with the PORV open would cause QT pressure to increase as the water temperature increased.

2nd part is correct. This is the calculated temperature for the stated conditions.

Basis for meeting the KA

This question matches the KA by requiring knowledge of expected PRT (Quench Tank) parameter changes during PORV testing.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References

PNS-PZR Obj: R17, Pg 32
PT/1/A/0201/004
Steam Tables

Student References Provided

SYS010 A3.01 - Pressurizer Pressure Control System (PZR PCS)
Ability to monitor automatic operation of the PZR PCS, including: (CFR: 41.7 / 45.5)
PRT temperature and pressure during PORV testing ....

401-9 Comments:

Remarks/Status

SYS012 A3.05 - Reactor Protection System (RPS)
Ability to monitor automatic operation of the RPS, including: (CFR: 41.7 / 45.5)
Single and multiple channel trip indicators ....

Given the following Unit 1 conditions:

- Reactor power = 100%
- 1A RPS Thot RTD power supply is lost

Which ONE of the following describes:

1) ALL RPS trips affected by the failure?

2) the actions preferred in accordance with OP/1/A/1105/014 (Control Room Instrumentation Operation And Information)?

A. 1. RCS High Outlet Temperature ONLY
2. Place MANUAL TRIP Keyswitch in "TRIP".

B. 1. RCS High Outlet Temperature ONLY
2. Place affected RPS Channel MANUAL BYPASS keyswitch in "BYP".

C. 1. RCS High Outlet Temperature and RCS Variable Low Pressure
2. Place MANUAL TRIP Keyswitch in "TRIP".

D. 1. RCS High Outlet Temperature and RCS Variable Low Pressure
2. Place affected RPS Channel MANUAL BYPASS keyswitch in "BYP".

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D

General Discussion

Answer A Discussion

Incorrect: First part is plausible since it is the only trip function in RPS with high temperature in its name.

Second part is incorrect because per OP/1/A/1105/014, Enclosure 4.7 for removal and restoration of RPS chanels, there is a note stateing that placing RPS channel in Manual Bypass is preferred to minimize risk of Reactor trip. It is plausible since it would be correct if this were a "required" RPS channel. However only 3 RPS channels are required IAW TS 3.3.1 and since there are no other conditions given, the channel with the failed NI would be considered not required.

Answer B Discussion

Incorrect: Incorrect. First part is plausible since it is the only trip function in RPS with high temperature in its name.

2nd part is correct. Per OP/1/A/1105/014, Enclosure 4.7 for removal and restoration of RPS chanels, there is a note stateing that placing RPS channel in Manual Bypass is preferred to minimize risk of Reactor trip.

Answer C Discussion

1st part is correct. The High Outlet Temperature trip uses Thot directly to determine if the trip setpoint has been reached. The Variable Low Pressure trip uses Thot in the formula to caculate the low pressure trip:
11.14Thot - 4706

Second part is incorrect because per OP/1/A/1105/014, Enclosure 4.7 for removal and restoration of RPS chanels, there is a note stateing that placing RPS channel in Manual Bypass is preferred to minimize risk of Reactor trip. It is plausible since it would be correct if this were a "required" RPS channel. However only 3 RPS channels are required IAW TS 3.3.1 and since there are no other conditions given, the channel with the failed NI would be considered not required.

Answer D Discussion

1st part is correct. The High Outlet Temperature trip uses Thot directly to determine if the trip setpoint has been reached. The Variable Low Pressure trip uses Thot in the formula to caculate the low pressure trip:
11.14Thot - 4706

2nd part is correct. Per OP/1/A/1105/014, Enclosure 4.7 for removal and restoration of RPS chanels, there is a note stateing that placing RPS channel in Manual Bypass is preferred to minimize risk of Reactor trip.

Basis for meeting the KA

Question matches the KA by requiring knowledge of the indications of RPS operation due to a component failure.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT40 Q38

Development References

ILT40 Q38
IC RPS, Obj: R6 Pg 43, 45
IC RCI pg 14
OP 1105 014 Encl 4.7

Student References Provided

SYS012 A3.05 - Reactor Protection System (RPS)
Ability to monitor automatic operation of the RPS, including: (CFR: 41.7 / 45.5)
Single and multiple channel trip indicators ....

401-9 Comments:

Remarks/Status

ILT 47 ONS SRO NRC Examination QUESTION 39 39 **D**

SYS013 2.2.39 - Engineered Safety Features Actuation System (ESFAS)
SYS013 GENERIC
Knowledge of less than or equal to one hour Technical Specification action statements for systems. (CFR: 41.7 / 41.10 / 43.2 / 45.13)

Given the following Unit 1 conditions:

- Reactor power = 50% stable
- Power to ES Channel 1 VOTERS is lost

1) The loss of power above ___(1)___ actuate ES Channel 1.

2) In accordance with TS 3.3.7 (ESPS Automatic Actuation Output Logic Channels) the Completion Time for placing the associated ES Ch 1 components in their ES configuration is ___(2)___.

Which ONE of the following completes the statements above?

A. 1. will
2. immediately

B. 1. will
2. one hour

C. 1. will NOT
2. immediately

D. 1. will NOT
2. one hour

ILT 47 ONS SRO NRC Examination QUESTION 39

39

D

General Discussion

Answer A Discussion

1st part is incorrect because the RO VOTER has to energize to initiate the ES channel. It is plausible because if it were RPS, it would be correct since an RPS channel does actuate when it loses power.

2nd part is incorrect because TS 3.3.7 allows up to 1 hour to put the components into their ES position or declare them INOPERABLE. It is plausible because there are numerous similar TS that require "immediate " action (CR indication, RPS instrumentation..).

Answer B Discussion

1st part is incorrect because the RO VOTER has to energize to initiate the ES channel. It is plausible because if it were RPS, it would be correct since an RPS channel does actuate when it loses power.

2nd part is correct. TS 3.3.7 allows up to 1 hour to declare components inoperable or place them in their ES configuration.

Answer C Discussion

1st part is correct. ES must energize to actuate.

2nd part is incorrect because TS 3.3.7 allows up to 1 hour to put the components into their ES position or declare them INOPERABLE. It is plausible because there are numerous similar TS that require "immediate " action (CR indication, RPS instrumentation..).

Answer D Discussion

1st part is correct. ES must energize to actuate.

2nd part is correct. TS 3.3.7 allows up to 1 hour to declare components inoperable or place them in their ES configuration.

Basis for meeting the KA

This question matches the KA by requiring knowledge of 1 hour TS associated with ESFAS.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

IC-ES Pg 54
TS 3.3.7
TS 3.3.7 B

Student References Provided

SYS013 2.2.39 - Engineered Safety Features Actuation System (ESFAS)
SYS013 GENERIC
Knowledge of less than or equal to one hour Technical Specification action statements for systems. (CFR: 41.7 / 41.10 / 43.2 / 45.13)

401-9 Comments:

Remarks/Status

SYS022 K1.01 - Containment Cooling System (CCS)

Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

SWS/cooling system ....

Given the following Unit 1 conditions:
Initial conditions:

- Time = 1200
- Reactor Power = 100%
- 1A MSLB inside the Reactor Building

Current conditions:

- Time = 1201
- Reactor Building Pressure = 3 psig increasing

Which ONE of the following describes the operation of 1A RBCU OUTLET, 1LPSW-18?

A. It is NORMALLY fully open however it will receive a signal to open from ES-5 at 1201

B. It is NORMALLY throttled and will go fully open when it receives a signal to open from ES-5 at 1201

C. It is NORMALLY fully open however it will receive a signal to open from ES-5 at 1204

D. It is NORMALLY throttled and will go fully open when it receives a signal to open from ES-5 at 1204

ILT 47 ONS SRO NRC Examination QUESTION 40

40

B

General Discussion

Answer A Discussion

Incorrect. The valve being fully open at 1200 is plausible since the associated RBCU inlet valve (1LPSW-16) normal position is fully open. ES-5 does send an open signal to 1LPSW-18 at 1201.

Answer B Discussion

Correct. The RBCU Cooler outlet valves are throttled during normal operation and go fully open when an ES signal is received. Since ES 5&6 actuate at 3 psig RB pressure, 1LPSW-18 will receive its open signal at 1201.

Answer C Discussion

Incorrect. The valve being fully open at 1200 is plausible since the associated RBCU inlet valve (1LPSW-16) normal position is fully open. Not receiving an open signal until 1204 is plausible since the start signal to the RBCU's is delayed for 3 minutes following ES 5&6 to ensure adequate bus voltages. Since the RBCU does not receive a start signal until 1204 it is plausible to believe that the associated LPSW outlet valve does not receive a signal to open until the RBCU has received a signal to start.

Answer D Discussion

Incorrect. The valve is throttled at 1200. Not receiving an open signal until 1204 is plausible since the start signal to the RBCU's is delayed for 3 minutes following ES 5&6 to ensure adequate bus voltages. Since the RBCU does not receive a start signal until 1204 it is plausible to believe that the associated LPSW outlet valve does not receive a signal to open until the RBCU has received a signal to start.

Basis for meeting the KA

Requires the ability to monitor Containment Cooling System valves (LPSW cooling water to RBCU's) for proper operation following an ES signal.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT44 (Q 66) NRC Exam

Development References
PNS-RBCPg 15, 16 ILT44 (Q66)

Student References Provided

SYS022 K1.01 - Containment Cooling System (CCS)

Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

SWS/cooling system ....

401-9 Comments:

Remarks/Status

SYS022 K2.02 - Containment Cooling System (CCS)
Knowledge of power supplies to the following: (CFR: 41.7)
Chillers ....

Which ONE of the following is the power supply for Reactor Building Cooling Unit (RBCU) 1A?

A. 1XS2

B. 1XS3

C. 1X8

D. 1X9

ILT 47 ONS SRO NRC Examination QUESTION 41

41

C

General Discussion

Answer A Discussion

Incorrect because the power supply for the A1 RBCU is 1X8. It is plausible because there are RBCU feeds from the 1XS load centers.

Answer B Discussion

Incorrect because the power supply for the A1 RBCU is 1X8. It is plausible because if it were the 1B RBCU, it would be correct.

Answer C Discussion

Correct.

Answer D Discussion

Incorrect because the power supply for the A1 RBCU is 1X8. It is plausible because if it were the 1C RBCU, it would be correct.

Basis for meeting the KA

Chief agreed that using RBCU's in place of chillers will meet this KA.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References
PNS-RBC Pg 18

Student References Provided

SYS022 K2.02 - Containment Cooling System (CCS)
Knowledge of power supplies to the following: (CFR: 41.7)
Chillers ....

401-9 Comments:

Remarks/Status

ILT 47 ONS SRO NRC Examination QUESTION 42 42 **C**

SYS026 K2.02 - Containment Spray System (CSS)
Knowledge of bus power supplies to the following: (CFR: 41.7)
MOVs ....

Which ONE of the following is the power supply to Building Spray Pump 1A discharge valve, 1BS-1?

A. 1XS-2

B. 1XS-3

C. 1XS-4

D. 1XS-5

General Discussion

Answer A Discussion

Incorrect because 1BS-1 is powered from 1XS-4. It is plausible because 1BS-4 is powered from 1XS-2

Answer B Discussion

Incorrect because 1BS-1 is powered from 1XS-4. It is plausible because other ECCS system valves (1LP-19) are powered from 1XS-3.

Answer C Discussion

Correct. 1BS-1 is powered from 1XS-4

Answer D Discussion

Incorrect because 1BS-1 is powered from 1XS-4. It is plausible because 1BS-2 is powered from 1XS-5

Basis for meeting the KA

This question matches the KA by requiring knowledge of the power supplies to Building Spray system valves.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	MODIFIED	

Development References
PNS-BS Pg 8, 9

Student References Provided

SYS026 K2.02 - Containment Spray System (CSS)
Knowledge of bus power supplies to the following: (CFR: 41.7)
MOVs ....

401-9 Comments:

Remarks/Status

SYS039 A1.10 - Main and Reheat Steam System (MRSS)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: (CFR: 41.5 / 45.5)

Air ejector PRM ....

Given the following Unit 1 conditions:

- Reactor power = 100%
- 1RIA-40 (CSAE Off-Gas Monitor) reading is rising slowly
- 1RIA-54 (Turbine Building (TB) Sump Monitor) is inoperable
- The operating crew has just entered AP/31 (Primary To Secondary Leakage) due to a 6 gpm leak in the 1A SG

1) In accordance with AP/31 an AO is required to __ (1) __.

2) Emergency Dose Limits __ (2) __ in affect.

A. 1. open and white tag the TB Sump Pump breakers
2. are

B. 1. open and white tag the TB Sump Pump breakers
2. are NOT

C. 1. align the TB Sump to the TB Sump Monitor Tanks
2. are

D. 1. align the TB Sump to the TB Sump Monitor Tanks
2. are NOT

ILT 47 ONS SRO NRC Examination QUESTION 43

43

B

General Discussion

Answer A Discussion

First part is correct. AP/31 directs the two turbine building sump pumps breaker's be white tagged and open.

Second part is incorrect and plausible. The emergency Dose limits are in effect on a SG tube leak only if the SGTR EOP is in effect. At 6 gpm the AP is used so normal dose limits apply. The EOP is entered at >25 gpm.

Answer B Discussion

First part is correct. AP/31 directs the two turbine building sump pumps breaker's be white tagged and open.

Second part is correct.. The Emergency Dose Limits are in effect on a SG tube leak only if the SGTR EOP is in effect. At 6 gpm the AP is used so normal dose limits apply. The EOP is entered at >25 gpm.

Answer C Discussion

First part is incorrect and plausible. 1104/048 TB Sump Operation directs that if TB Sump sample results activity > 10 EC, TB Sump must be pumped to TB Sump Monitor Tanks

Second part is incorrect and plausible. The emergency Dose limits are in effec

Answer D Discussion

First part is incorrect and plausible. 1104/048 TB Sump Operation directs that if TB Sump sample results activity > 10 EC, TB Sump must be pumped to TB Sump Monitor Tanks

Second part is correct.. The emergency Dose limits are in effect

Basis for meeting the KA

Question requires knowledge of the process during a tube leak to ensure an unmonitored release does not occur (exceed limits). This is consistent with the ALARA goals. The distinction between normal and emergency dose limits is tested for knowledge of EOP/AP as it relates to leak size.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT39 Q63

Development References

AP 31
EAP-APG
SGTR PA Page
ILT39 Q63

Student References Provided

SYS039 A1.10 - Main and Reheat Steam System (MRSS)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: (CFR: 41.5 / 45.5)

Air ejector PRM ....

401-9 Comments:

Remarks/Status

SYS039 K5.01 - Main and Reheat Steam System (MRSS)
Knowledge of the operational implications of the following concepts as the apply to the MRSS: (CFR: 441.5 / 45.7)
Definition and causes of steam/water hammer ....

Given the following Unit 3 conditions:

- Unit startup in progress
- Turbine heatup has been initiated
- Turbine Bypass Lines Pumping Trap has malfunctioned and is not removing moisture

1) Plant damage, as a result of the malfunctioning pumping trap is a concern due to the potential of__ (1) __.

2) The Turbine Bypass Lines Pumping Trap is aligned to the __ (2) __.

Which ONE of the following completes the statements above?

A. 1. a water hammer
2. Condenser

B. 1. a water hammer
2. Condensate Storage Tank

C. 1. moisture impingement on the turbine blades
2. Condenser

D. 1. moisture impingement on the turbine blades
2. Condensate Storage Tank

General Discussion

Answer A Discussion

1st part is correct. A water hammer could result.

2nd part is correct. The pumping trap is aligned to the condenser on U3.

Answer B Discussion

1st part is correct. A water hammer could result.

2nd part is incorrect because they are aligned to the condenser. It is plausible because other steam condensate drains are aligned to the CST. Ex. CSAE after cooler
drains, Steam Packing exhauster condenser drains, etc.

Answer C Discussion

First part is incorrect because the concern is water hammer. It is plausible because on Unit 1 it would be correct because the MS lines use the Turbine Bypass Line Pumping Trap.

2nd part is correct. The pumping trap is aligned to the condenser on U3.

Answer D Discussion

First part is incorrect because the concern is water hammer. It is plausible because on Unit 1 it would be correct because the MS lines use the Turbine Bypass Line Pumping Trap.

2nd part is incorrect because they are aligned to the condenser. It is plausible because other steam condensate drains are aligned to the CST. Ex. CSAE after cooler
drains, Steam Packing exhauster condenser drains, etc.

Basis for meeting the KA

Question requires knowledge of the operational implications of water hammer that could result from a broke pumping trap.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT41 (Q 41) NRC Exam

Development References

STG-MSPg 14, 15
ILT41 (Q41)

Student References Provided

SYS039 K5.01 - Main and Reheat Steam System (MRSS)

Knowledge of the operational implications of the following concepts as the apply to the MRSS: (CFR: 441.5 / 45.7)

Definition and causes of steam/water hammer ....

401-9 Comments:

Remarks/Status

ILT 47 ONS SRO NRC Examination QUESTION 45 45 **D**

SYS059 K1.07 - Main Feedwater (MFW) System

Knowledge of the physical connections and/or cause-effect relationships between the MFW and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

ICS ....

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 40%
- Loop B FDW valve ΔP selected to 1B2

Current conditions:

- 1B2 Loop B FDW valve ΔP fails LOW

1) Feedwater Flow will initially __ (1) __.

2) AP/28 (ICS Instrument Failures) will ensure BOTH __ (2) __ are in HAND to stabilize the plant.

Which ONE of the following completes the statements above?

A. 1. decrease
2. FDW Masters

B. 1. decrease
2. Main FDW Pumps

C. 1. increase
2. FDW Masters

D. 1. increase
2. Main FDW Pumps

ILT 47 ONS SRO NRC Examination QUESTION 45 45 **D**

General Discussion

Answer A Discussion

First part is incorrect because flow wil linitially increase due to MFP speed increasing. It is plausible because the control valves will decrease FDW flow after the FDW pumps initially increase flow.

Second part is incorrect because the MFWP baileys will still change FDW flow. It is plausible because both FDW masters are normally taken to hand during PTR.

Answer B Discussion

First part is incorrect because flow wil linitially increase due to MFP speed increasing. It is plausible because the control valves will decrease FDW flow after the FDW pumps initially increase flow.

2nd part is correct. MFW pumps have to be taken to HAND because they will still adjust FDWP speed even if the FDW Masters are taken to HAND.

Answer C Discussion

1st part is correct. As speed increases, FDW flow will increase initially.

Second part is incorrect because the MFWP baileys will still change FDW flow. It is plausible because both FDW masters are normally taken to hand during PTR.

Answer D Discussion

1st part is correct. As speed increases, FDW flow will increase initially.

2nd part is correct. MFW pumps have to be taken to HAND because they will still adjust FDWP speed even if the FDW Masters are taken to HAND..

Basis for meeting the KA

Question matches the KA by requiring knowledge of how an ICS malfunction affects FDW components.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT42 Q59

Development References

STG-ICS Ch 4, Obj: R20 Pg 25
ILT42 Q59

Student References Provided

SYS059 K1.07 - Main Feedwater (MFW) System
Knowledge of the physical connections and/or cause-effect relationships between the MFW and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
ICS ....

401-9 Comments:

Remarks/Status

SYS059 K3.03 - Main Feedwater (MFW) System
Knowledge of the effect that a loss or malfunction of the MFW will have on the following: (CFR: 41.7 / 45.6)
S/GS ....

Given the following Unit 1 conditions:
Initial conditions:

- Reactor power = 100%

Current conditions:

- Condenser vacuum = 18.5 inches Hg stable
- 1TA and 1TB de-energized

SG levels will be automatically controlled at ________.

Which ONE of the following completes the statement above?

A. 25 inches SUR

B. 30 inches XSUR

C. 50% OR

D. 240 inches XSUR

ILT 47 ONS SRO NRC Examination QUESTION 46

46

D

General Discussion

Answer A Discussion

Incorrect. Plausible because it would be correct if on Main FDW with RCPs.

Answer B Discussion

Incorrect. Plausible because it would be correct if on EFDW with RCPs.

Answer C Discussion

Incorrect. Plausible because it would be correct if on Main FDW without RCPs.

Answer D Discussion

Correct. At 19 inches Hg Main FDW will trip. Without 1TA and 1TB (no RCPs) EFDW will control SG level at 240 inches XSUR.

Basis for meeting the KA

Question requires knowledge of how a malfunction of the MFW system (loss of MFW) will have on SG level.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT42 (Q 27) NRC Exam

Development References

CF-EF Pg 27
CF-FPT Pg 30
ILT42 Q27

Student References Provided

SYS059 K3.03 - Main Feedwater (MFW) System

Knowledge of the effect that a loss or malfunction of the MFW will have on the following: (CFR: 41.7 / 45.6)

S/GS ....

401-9 Comments:

Remarks/Status

ILT 47 ONS SRO NRC Examination **QUESTION** **47** 47 **D**

SYS061 K6.02 - Auxiliary / Emergency Feedwater (AFW) System

Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: (CFR: 41.7 / 45.7)

Pumps ....

Given the following Unit 1 conditions:
Initial conditions:

- Reactor power = 100%
- Unit 1 TDEFWP unavailable

Current conditions:

- Both Main FDW pumps trip
- 1B MDEFWP fails to start

Which ONE of the following describes actions directed by the EOP to remove core decay heat?

Initiate…

A. Rule 3 (Loss of Main or Emergency Feedwater) and cross connect with an alternate unit to supply the 1B Steam Generator

B. Rule 3 (Loss of Main or Emergency Feedwater) to decrease SG pressure and feed with Condensate Booster pumps

C. Rule 4 (Initiation of HPI Forced Cooling) if RCS pressure reaches 2300 psig

D. EOP Encl. 5.9 (Extended EFDW Operation) and feed both SG's with 1A MDEFWP

ILT 47 ONS SRO NRC Examination QUESTION 47

47

D

General Discussion

Answer A Discussion

Incorrect: Plausible since cross connecting with alternate unit is a mitigation strategy utilized by Rule 3 however it is applied if no EFDWPs are available on the subject unit.

Answer B Discussion

Incorrect: Plausible since CBP feed is a strategy utilized by Rule 3 and it would be correct if the 1A MDEFWP had also been lost.

Answer C Discussion

Incorrect: Plausible since HPI FC is utilized as a strategy in RULE 4 and would be correct if the 1A MDEFWP had also been lost since it is only applied if neither SG can be fed and RCS pressure reached 2300 psi

Answer D Discussion

CORRECT: If only one MDEFWP is available Rule 3 will send you to Encl. 5.9 which will direct opening FDW-313 & 314 and feeding both SG’s with one MDEFWP.

Basis for meeting the KA

Requires knowledge of how AFW components are utilized based on loss of MFDWP’s, TDEFWP, and one MDEFWP

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	2010A (Q46) NRC Exam

Development References

EOP Rule 3
EOP Encl 5.9
EOP-LOHT Pg 12
2010A Q46

Student References Provided

SYS061 K6.02 - Auxiliary / Emergency Feedwater (AFW) System

Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: (CFR: 41.7 / 45.7)

Pumps ....

401-9 Comments:

Remarks/Status

SYS062 A4.01 - AC Electrical Distribution System
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 / to 45.8)
All breakers (including available switchyard) ....

Given the following plant conditions:

- ACB-2 (Keowee 2 Generator BKR) CLOSED
- ACB-3 (Keowee 1 Emergency Feeder BKR) CLOSED
- A LOOP (Switchyard Isolation) causes ALL 4160 V switchgear (1TC, 1TD, and 1TE) to de-energize.

Which ONE of the following describes the response of Keowee switchgear power supplies?

A. 1X switchgear de-energizes and then is restored 15 seconds later

B. 2X switchgear de-energizes and then is restored 36 seconds later

C. 1X switchgear de-energizes and MUST be restored manually

D. 2X switchgear de-energizes and MUST be restored manually

ILT 47 ONS SRO NRC Examination QUESTION 48

48

A

General Discussion

Answer A Discussion

Correct. 1X will lose power because 1TC is de-energized due to the LOOP. After 15 seconds 1TC will regain power from Keowee Unit 2 and ACB-7 will reclose powering 1X swichgear.

Answer B Discussion

Incorrect. Pl;ausible because would be correct without the LOOP.

Answer C Discussion

Incorrect. Plausible because would be correct if breakers were in manual.

Answer D Discussion

Incorrect. Plausible because would be correct if breakers were in manual.

Basis for meeting the KA

This question matches the KA by requiring knowledge of AC Electrical Distribution breakers.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT43 (Q 46) NRC Exam

Development References

EL-KHG Pg 35
ILT43 Q46
EL PSL Pg 48
KHU Dwg

Student References Provided

SYS062 A4.01 - AC Electrical Distribution System

Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 / to 45.8)

All breakers (including available switchyard) ....

401-9 Comments:

Remarks/Status

SYS063 A2.01 - DC Electrical Distribution System

Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Grounds ....

Given the following Unit 1 conditions:

- Reactor power = 100%
- 1SA-04/E-6 (125 Volt Ground Trouble) actuates

1) 1SA-04/E-6 ARG directs the Operator to __ (1) __ to determine if the ground is on the battery or the Bus.

2) 1SA-04/E-6 actuating indicates that the ground is located on __ (2) __.

Which ONE of the following completes the statements above?

A. 1. rotate the Ground Relay Selector Switch
2. Unit 1 ONLY

B. 1. rotate the Ground Relay Selector Switch
2. any Unit

C. 1. isolate the battery from the Bus
2. Unit 1 ONLY

D. 1. isolate the battery from the Bus
2. any Unit

General Discussion

Answer A Discussion

Incorrect.
First part is incorrect and plausible. Plausible as operation of this switch is addressed in the ARG however its purpose is to be used for testing of the ground lamp circuits

Second part is incorrect and plausible. The alarm test lights are on Unit 1. An operator could reasonably conclude that an alarm is Unit specfic since each unit has a ground trouble Statalarm.

Answer B Discussion

Incorrect.
First part is incorrect and plausible. Plausible as operation of this switch is addressed in the ARG however its purpose is to be used for testing of the ground lamp circuits

Second part is correct. There is only one ground detection system. It is shared by all three units. The statalrm cannot be used to determine which unit is affected as all three units are normally cross connected.

Answer C Discussion

Incorrect.
First part is correct. The ARG directs isolating the battery from the bus to determine if the ground in on the battery or the bus.

Second part is incorrect and plausible. The alarm test lights are on Unit 1. An operator could reasonably conclude that an alarm is Unit specfic since each unit has a ground trouble Statalarm.

Answer D Discussion

Correct.
First part is correct. The ARG directs isolating the battery from the bus to determine if the ground in on the battery or the bus.

Second part is correct. There is only one ground detection system. It is shared by all three units. The statalrm cannot be used to determine which unit is affected as all three units are normally cross connected.

Basis for meeting the KA

Question requires knowledge of actions contained in Alarm Response procedures for detecting grounds and impact of those actions on the plant.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT39 (Q 13) NRC Exam

Development References

EL-DCD Pg 24, 30
1SA-04/E-6
ILT39 Q13

Student References Provided

SYS063 A2.01 - DC Electrical Distribution System
Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)
Grounds ....

401-9 Comments:

Remarks/Status

ILT 47 ONS SRO NRC Examination QUESTION 50 50 **B**

SYS064 A3.13 - Emergency Diesel Generator (ED/G) System
Ability to monitor automatic operation of the ED/G system, including: (CFR: 41.7 / 45.5)
Rpm controller/megawatt load control (breaker-open/breaker-closed effects) ....

Given the following plant conditions:

- Operators are preparing to synchronize KHU-2 to the grid
- OP/0/A/1106/019, (Keowee Hydro At Oconee) in progress
- Grid Frequency = 59.9 cycles
- Keowee Frequency = 60.3 cycles
- Keowee 2 Line Volts = 13.7 kV
- Keowee 2 Output Volts = 15.2 kV

1) KHU Unit 2 __ (1) __ will be used to adjust the synchroscope indication.

2) If ACB-2 is closed with the above indications, generator MVARs will be __ (2) _.

Which ONE of the following completes the statements above?

A. 1. Auto Voltage Adjuster
2. positive

B. 1. Speed Changer Motor
2. positive

C. 1. Auto Voltage Adjuster
2. negative

D. 1. Speed Changer Motor
2. negative

General Discussion

Answer A Discussion

Incorrect:
First part is incorrect and plausible. The voltage regulator (AVA) and the generator load/speed control are the two primary controls for the Keowee unit. It is reasonable that the candidate will confuse the two control devices and determine the AVA is used to adjust the synchroscope.

Second part is correct. Generator output voltage is greater than Line volts which will cause MVARs to be positive.

Answer B Discussion

Correct:
First part is correct. Keowee frequency is higher than the grid so synchroscope will be spinning clockwise which will require use of the Unit 2 Speed Changer motor to lower the Keowee generator frequency.

Second part is correct. Generator output voltage is greater than Line volts which will cause MVARs to be positive.
Answer C

Answer C Discussion

Incorrect:
First part is incorrect and plausible. The voltage regulator (AVA) and the generator load/speed control are the two primary controls for the Keowee unit. It is reasonable that the candidate will confuse the two control devices and determine the AVA is used to adjust the synchroscope.

Second part is incorrect and plausible. It is reasonable that the candidate not recognize the direction the voltage missmatch is in and determine negative MVARs will be generated.

Answer D Discussion

Incorrect.
First part is correct. Keowee frequency is higher than the grid so synchroscope will be spinning clockwise which will require use of the Unit 2 Speed Changer motor to lower the Keowee generator frequency.

Second part is incorrect and plausible. It is reasonable that the candidate not recognize the direction the voltage missmatch is in and determine negative MVARs will be generated.

Basis for meeting the KA

Question matches the KA by requiring knowledge of KHU controllers and how MVARs will adjust when closing the output breaker.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT39 (Q 49) NRC Exam

Development References

EL-KHG Pg 17
ILT39 Q49

Student References Provided

SYS064 A3.13 - Emergency Diesel Generator (ED/G) System
Ability to monitor automatic operation of the ED/G system, including: (CFR: 41.7 / 45.5)
Rpm controller/megawatt load control (breaker-open/breaker-closed effects) ....

401-9 Comments:

Remarks/Status

SYS073 A1.01 - Process Radiation Monitoring (PRM) System

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM system controls including: (CFR: 41.5 / 45.7)

Radiation levels ....

Given the following Unit 1 conditions:

- Reactor in MODE 5
- RB Purge is in progress
- Reactor Building Airborne activity is increasing .

RM Reactor BLDG Purge Disch RAD Inhibit will occur ____(1)____ and will close ____(2)____.

Which ONE of the following completes the statement above?

A. 1. prior to the "switchover" from 1RIA-45 to 1RIA-46 occurring
2. 1PR-2 through 1PR-5 ONLY

B. 1. prior to the "switchover" from 1RIA-45 to 1RIA-46 occurring
2. 1PR-1 through 1PR-6

C. 1. after the "switchover" from 1RIA-45 to 1RIA-46 occurs.
2. 1PR-2 through 1PR-5 ONLY

D. 1. after the "switchover" from 1RIA-45 to 1RIA-46 occurs
2. 1PR-1 through 1PR-6

General Discussion

Answer A Discussion

1st part correct. Both RIA-45 and RIA-46 will cause the isolation however, with no equipment failures, RIA-45 will cause the isolation before the switchover to RIA-46 occurs.

2nd part is correct. RIA-45 will only cause PR-2 through PR-5 to isolate.

Answer B Discussion

1st part correct. Both RIA-45 and RIA-46 will cause the isolation however, with no equipment failures, RIA-45 will cause the isolation before the switchover to RIA-46 occurs.

2nd part is incorrect because 1PR-1 and 1PR-6 do not close when the alarm is received. It is plausible because 1PR-1 through 1PR-6 do isolate on an ES signal.

Answer C Discussion

1st part is incorrect because the isolation will occur will occur before the switchover occurs. It is plausible because RIA-46 does perform the same function as RIA-45 and the switchover will occur as levels continue to rise. RIA-46 causing the isolation is a backup to RIA-45 in case it fails to isolate RB Purge.

2nd part is correct. RIA-45 will only cause PR-2 through PR-5 to isolate.

Answer D Discussion

1st part is incorrect because the isolation will occur will occur before the switchover occurs. It is plausible because RIA-46 does perform the same function as RIA-45 and the switchover will occur as levels continue to rise. RIA-46 causing the isolation is a backup to RIA-45 in case it fails to isolate RB Purge.

2nd part is incorrect because 1PR-1 and 1PR-6 do not close when the alarm is received. It is plausible because 1PR-1 through 1PR-6 do isolate on an ES signal

Basis for meeting the KA

This question matches the KA by requiring the ability to predict the system response due to increasing radiation levels (in order to prevent exceeding limits at the site boundary).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References
RAD RIA Pg 23

Student References Provided

SYS073 A1.01 - Process Radiation Monitoring (PRM) System

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM system controls including: (CFR: 41.5 / 45.7)

Radiation levels ....

401-9 Comments:

Remarks/Status

SYS061 K4.06 - Auxiliary / Emergency Feedwater (AFW) System
Knowledge of AFW design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)
AFW startup permissives ....

Given the following Unit 3 conditions:
Time = 1200

- Reactor power = 100%
- 3B MDEFWP switch in “AUTO 2”
- 3A MDEFWP switch in “AUTO 1” for testing

Time = 1201

- BOTH Main Feedwater pumps trip
- 3MS-87 (MS to TDEFDWP Control) fails closed

1) The 3A MD EFDW pump ____(1)____ be operating.

2) The TD EFDW pump ____(2)____ be operating.

Which ONE of the following completes the statements above at time = 1202 assuming NO operator actions?

A. 1. will
2. will

B. 1. will
2. will NOT

C. 1. will NOT
2. will

D. 1. will NOT
2. will NOT

General Discussion

Answer A Discussion

1st part is incorrect because the 3A MD EFDW pump will not be operating because the FDWP Hyd Oil Press start logic is not active when in Auto-1. It is plausible because if the switch were in Auto-2, it would be correct.

2nd part is correct. Aux steam will automatically provide steam to the TD EFDW pump.

Answer B Discussion

1st part is incorrect because the 3A MD EFDW pump will not be operating because the FDWP Hyd Oil Press start logic is not active when in Auto-1. It is plausible because if the switch were in Auto-2, it would be correct.

2nd part is incorrect because the TD EFDWP will be operating with steam coming from the Aux Steam System. It is plausible because the MS supply valve has failed closed.

Answer C Discussion

1st part is correct. The A MD EFDWP will not start because when in Auto-1, only the Dryout protection logic is active.

2nd part is correct. Aux steam will automatically provide steam to the TD EFDW pump.

Answer D Discussion

1st part is correct. The A MD EFDWP will not start because when in Auto-1, only the Dryout protection logic is active.

2nd part is incorrect because the TD EFDWP will be operating with steam coming from the Aux Steam System. It is plausible because the MS supply valve has failed closed.

Basis for meeting the KA

This question matches the KA by requiring knowledge of EFW pump starting interlocks.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	ILT40 Q45

Development References

ILT40 Q45
EFW Initiation Logic
CF EF Pg 17
STG AS Pg 9

Student References Provided

SYS061 K4.06 - Auxiliary / Emergency Feedwater (AFW) System
Knowledge of AFW design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)
AFW startup permissives ....

401-9 Comments:

Remarks/Status

SYS076 K4.03 - Service Water System (SWS)

Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following: (CFR: 41/7)

Automatic opening features associated with SWS isolation valves to CCW heat exchanges ....

During normal operation of the CC system…

1) CC flow through each letdown cooler is maintained at ____(1)____ gpm.

2) If letdown flow were increased, CC outlet temperature on the in-service CC cooler would be maintained by ____(2)____ operation of the associated LPSW valve.

Which ONE of the following completes the statements above?

A. 1. 200 gpm
2. manual

B. 1. 200 gpm
2. automatic

C. 1. 400 gpm
2. manual

D. 1. 400 gpm
2. automatic

ILT 47 ONS SRO NRC Examination QUESTION 53

53

B

General Discussion

Answer A Discussion

1st part is correct. Both LD coolers are throttled to 200 gpm each during normal operation.

2nd part is incorrect since the LPSW controller is an automatic control valve which controls at setpoint. It is plausibe because the CC valves are all adjusted manually.

Answer B Discussion

1st part is correct. Both LD coolers are throttled to 200 gpm each during normal operation.

2nd part is correct. LPSW controller is an automatic control valve which controls at setpoint

Answer C Discussion

1st part is incorrect because the flow through each cooler is 200 gpm. It is plausible because the total flow through the LD coolers is 400 gpm. This flow is set up during system startup.

2nd part is incorrect since the LPSW controller is an automatic control valve which controls at setpoint. It is plausibe because the CC valves are all adjusted manually.

Answer D Discussion

1st part is incorrect because the flow through each cooler is 200 gpm. It is plausible because the total flow through the LD coolers is 400 gpm. This flow is set up during system startup.

2nd part is correct. LPSW controller is an automatic control valve which controls at setpoint

Basis for meeting the KA

This question matches the KA by requiring knowledge of LPSW valve (CC cooler outlet) automatically repositions (in the open direction) to maintain CC temperature.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References
PNS CC Pg 10, 11

Student References Provided

SYS076 K4.03 - Service Water System (SWS)

Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following: (CFR: 41/7)

Automatic opening features associated with SWS isolation valves to CCW heat exchanges ....

401-9 Comments:

Remarks/Status

ILT 47 ONS SRO NRC Examination **QUESTION** **54** 54 **C**

SYS078 2.4.2 - Instrument Air System (IAS)
SYS078 GENERIC
Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions. (CFR: 41.7 / 45.7 / 45.8)

Given the following Unit 1 conditions:

- Reactor power = 100%
- Instrument Air Pressure decreasing
- AP/22 (Loss of Instrument Air) initiated

Current conditions:

- Instrument Air pressure = 61 psig decreasing
- FDW Pump delta P OAC alarms actuate
- 1A & 1B Main FDW Pump speeds are both increasing slowly

Which ONE of the following describes the actions required by AP/22?

A. Commence a plant shutdown. If at any time two or more CRD temperatures are >180ºF, then trip the reactor.

B. Commence a plant shutdown. If at any time SG level approaches main FDW pump trip criteria, then trip the reactor.

C. Manually trip the reactor. Manually trip both main FDW pumps.

D. Manually trip the reactor. Take both FDW Masters to Hand and decrease demand to zero.

ILT 47 ONS SRO NRC Examination QUESTION 54 54 **C**

General Discussion

Answer A Discussion

Incorrect because AP/22 does not direct a plant shutdown (other than a trip). Plausible because with IA pressure decreasing, the unit will not be able to stay at power as components are not able to be controlled. This is the immediate trip criteria for CRDM's and would be applicable in this condition.

Answer B Discussion

Incorrect because AP/22 does not direct a plant shutdown (other than a trip). Plausible because with IA pressure decreasing, the unit will not be able to stay at power as components are not able to be controlled. It is plausible in that OMP 1-18 dictates a Manual Rx Trip and tripping of both MFWPS if any SG reaches >96% on the OR level.

Answer C Discussion

Correct. AP/22 requires the reactor to be tripped and then MFW pumps to be tripped when IA pressure is < 65 psig when in Mode 1 or 2.

Answer D Discussion

Incorrect because AP/22 requires MFDW pumps to be tripped immediately after the reactor is manually due to loss of FDW controllability. Plausible in that the candidate could erroneously think that Feedwater control valves (and FDW demand) would still be controllable if taken to Hand on the ICS stations.

Basis for meeting the KA

Question matches the KA by requiring knowledge of when to enter the EOP based on system conditions.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT42 (Q 15) NRC Exam

Development References

SSS-IA Pg 47
AP/22
IA Composite
ILT42 Q15

Student References Provided

SYS078 2.4.2 - Instrument Air System (IAS)
SYS078 GENERIC
Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions. (CFR: 41.7 / 45.7 / 45.8)

401-9 Comments:

Remarks/Status

SYS103 K3.03 - Containment System

Knowledge of the effect that a loss or malfunction of the containment system will have on the following: (CFR: 41.7 / 45.6)

Loss of containment integrity under refueling operations. ....

Given the following Unit 1 conditions:
Time = 0805

- Reactor in MODE 6
- Fuel offload is in progress
- Reactor Building Normal Sump (RBNS) is being pumped
- A fuel assembly is dropped

Time = 0809

- A High Radiation Annunciator in the Control Room alarms
- The Reactor Building Normal Sump has failed to isolate
- AP/9 SPENT FUEL DAMAGE is initiated

1) 1RIA __(1)__ in HIGH alarm should have caused the RBNS isolation.

2) If the RB Normal sump isolation valves are the last open penetrations to be closed and are closed at 0830, the criteria for isolating open penetrations per AP/9 __(2)__ been met.

Which ONE of the following completes the statements above?

A. 1. 4 (Reactor Building Entrance)
2. has

B. 1. 4 (Reactor Building Entrance)
2. has NOT

C. 1. 49 (RB Gas)
2. has

D. 1. 49 (RB Gas)
2. has NOT

ILT 47 ONS SRO NRC Examination QUESTION 55

55

C

General Discussion

Answer A Discussion

1st part incorrect because RIA-4 in alarm will not isolate the RB Sump. It is plausible because, like RIA-49, it will cause a RB Evacuation alarm.

2nd part is correct because the the 30 minute criteria (as stated in a NOTE in AP 9) for isolating any open penetrations has been met.

Answer B Discussion

1st part incorrect because RIA-4 in alarm will not isolate the RB Sump. It is plausible because, like RIA-49, it will cause a RB Evacuation alarm.

2nd part is incorrect because the 30 minute criteria stated in AP/9 for isolating open penetrations has been met. It is plausible because if it were at 0836, it would be correct.

Answer C Discussion

1st part is correct. 1RIA-49 will sound the RB Evacuation alarm and isolate the RB sump.

2nd part is correct because the the 30 minute criteria (as stated in a NOTE in AP 9) for isolating any open penetrations has been met.

Answer D Discussion

1st part is correct. 1RIA-49 will sound the RB Evacuation alarm and isolate the RB sump.

2nd part is incorrect because the 30 minute criteria stated in AP/9 for isolating open penetrations has been met. It is plausible because if it were at 0836, it would be correct.

Basis for meeting the KA

This question matches the KA by requiring knowledge of establishing containment integrity in the event of a malfunction (hi rad).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	MODIFIED	ILT43 (Q 23) NRC Exam

Development References

RAD-RIA Pg 23
AP/9
ILT43 Q23

Student References Provided

SYS103 K3.03 - Containment System

Knowledge of the effect that a loss or malfunction of the containment system will have on the following: (CFR: 41.7 / 45.6)

Loss of containment integrity under refueling operations. ....

401-9 Comments:

Remarks/Status

SYS002 K6.07 - Reactor Coolant System (RCS)

Knowledge of the effect or a loss or malfunction on the following RCS components: (CFR: 41.7 / 45.7)

Pumps ....

Given the following Unit 1 conditions:

- Reactor power = 80%
- 1B1 RCP trips

1) The ICS will initiate a unit runback at __ (1) __%/minute.

2) When the runback is complete, reactor power will be approximately __ (2) _%.

Which ONE of the following completes the statements above?

A. 1. 20
   2. 65

B. 1. 20
   2. 74

C. 1. 25
   2. 65

D. 1. 25
   2. 74

ILT 47 ONS SRO NRC Examination QUESTION 56

56

D

General Discussion

Could part two be the power level you run back to? 65% vs 74%

Answer A Discussion

1st part is incorrect because the reactor will run back at 25% per minute. It is plausible because if it were a loss of RC Flow (measured by actual loop flow as opposed RCP breaker position), it would be correct.

2nd part is incorrect because for a RCP breaker trip, power will run back to 74%. It is plausible because if it were running back due to a MFWP trip, it would be correct.

Answer B Discussion

1st part is incorrect because the reactor will run back at 25% per minute. It is plausible because if it were a loss of RC Flow (measured by actual loop flow as opposed RCP breaker position), it would be correct.

2nd part is correct. When a RCP breaker trips, ICS will run the plant back to ~ 74% power.

Answer C Discussion

1st part is correct. When a RCP trips, ICS will run back the plant at 25% per minute.

2nd part is incorrect because for a RCP breaker trip, power will run back to 74%. It is plausible because if it were running back due to a MFWP trip, it would be correct.

Answer D Discussion

1st part is correct. When a RCP trips, ICS will run back the plant at 25% per minute.

2nd part is correct. When a RCP breaker trips, ICS will run the plant back to ~ 74% power.

Basis for meeting the KA

This question matches the KA by requiring knowledge of how the impact of one RCP tripping will effect the RCS.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT41 (Q 57) NRC Exam

Development References

STG-ICS Ch2 Pg 29
PNS-RCS Pg 12
ILT41 Q57

Student References Provided

SYS002 K6.07 - Reactor Coolant System (RCS)

Knowledge of the effect or a loss or malfunction on the following RCS components: (CFR: 41.7 / 45.7)

Pumps ....

401-9 Comments:

Remarks/Status

SYS016 K5.01 - Non-Nuclear Instrumentation System (NNIS)
Knowledge of the operational implication of the following concepts as they apply to the NNIS: (CFR: 41.5 / 45.7)
Separation of control and protection circuits ....

Given the following Unit 1 conditions:
Time = 0800

- Reactor power = 100%
- NR RCS pressure Channel B has failed high

Time = 0801

- NR RCS pressure Channel E fails high

1) 1RC-66 (PORV) ____(1)____fail open.

2) The reactor ____(2)____ receive a "High RCS Pressure" trip signal.

Based on the plant conditions at 0801, complete the above statements.
(Assume NO operator actions)

A. 1. will
2. will

B. 1. will
2. will NOT

C. 1. will NOT
2. will

D. 1. will NOT
2. will NOT

General Discussion

Answer A Discussion

1st part is correct. NR channels A, B and E feed a median select circuit. If 2 of the 3 signals fail high, one of the high signals will pass on to the control circuits including the PORV. This will cause the PORV to fail open.

2nd part is incorrect because the reactor will NOT receive a HP trip because NR Channel E does not feed RPS. Therefore there is only a 1/4 signal to initiate a reactor trip when 2/4 is required. It is plausible because 1) any combination of NR Channels A, B, C and D would be correct and 2) the reactor will trip on LOW RCS pressure due to the PORV being open.

If NR RCS Pressure Channels A & B failed high, this answer would be correct.

Answer B Discussion

1st part is correct. NR channels A, B and E feed a median select circuit. If 2 of the 3 signals fail high, one of the high signals will pass on to the control circuits including the PORV. This will cause the PORV to fail open.

2nd part is correct. NR RCS pressure channel E does not feed RPS, therefore the reactor will only receive a trip signal on 1/4 channels when 2/4 is required.

Answer C Discussion

1st part is incorrect because the PORV will fail open. It is plausible because 3 of the 5 RPS channels feed the control circuit (A, B and E). If one of the failed channels were NR channels C or D, it would be correct.

2nd part is incorrect because the reactor will NOT receive a HP trip because NR Channel E does not feed RPS. Therefore there is only a 1/4 signal to initiate a reactor trip when 2/4 is required. It is plausible because 1) any combination of NR Channels A, B, C and D would be correct and 2) the reactor will trip on LOW RCS pressure due to the PORV being open.

If NR RCS Pressure Channels A & C failed high, this answer would be correct.

Answer D Discussion

1st part is incorrect because the PORV will fail open. It is plausible because 3 of the 5 RPS channels feed the control circuit (A, B and E). If one of the failed channels were NR channels C or D, it would be correct.

2nd part is correct. NR RCS pressure channel E does not feed RPS, therefore the reactor will only receive a trip signal on 1/4 channels when 2/4 is required.

If NR RCS Pressure Channels C & E failed high, this answer would be correct.

Basis for meeting the KA

This question matches the KA by requiring knowledge of how the NR RCS pressure signals feed the control and protections circuits (how they are seperated).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References
IC-RCI Pg 29, 30 RCI RCS Pressure Dwg

Student References Provided

SYS016 K5.01 - Non-Nuclear Instrumentation System (NNIS)

ILT 47 ONS SRO NRC Examination QUESTION 57 57 **B**

Knowledge of the operational implication of the following concepts as they apply to the NNIS: (CFR: 41.5 / 45.7)
Separation of control and protection circuits ....

401-9 Comments:	Remarks/Status

ILT 47 ONS SRO NRC Examination QUESTION 58 58 **D**

SYS075 K2.03 - Circulating Water System
Knowledge of bus power supplies to the following: (CFR: 41.7)
Emergency/essential SWS pumps ....

The C LPSW Pump is normally powered from __(1)__ and it __(2)__ have an alternate supply from another unit.

A. 1. 1TC
2. does

B. 1. 1TC
2. does NOT

C. 1. 2TC
2. does

D. 1. 2TC
2. does NOT

ILT 47 ONS SRO NRC Examination QUESTION 58

58

D

General Discussion

Answer A Discussion

1st part is incorrect because the C LPSW power supply is 2TC. It is plausible because both the A and B LPSWP's normal power supply is from 1TC.

2nd part is incorrect because the C LPSW does not have an alternate power supply. It is plausible because if it were the B LPSW pump, it would be correct.

Answer B Discussion

1st part is incorrect because the C LPSW power supply is 2TC. It is plausible because both the A and B LPSWP's normal power supply is from 1TC.

2nd part is correct. The C LPSW does not have an alternate power supply.

Answer C Discussion

1st part is correct. 2TC is the power supply to the C LPSW pump.

2nd part is incorrect because the C LPSW does not have an alternate power supply. It is plausible because if it were the B LPSW pump, it would be correct.

Answer D Discussion

1st part is correct. 2TC is the power supply to the C LPSW pump.

2nd part is correct. The C LPSW does not have an alternate power supply.

Basis for meeting the KA

This question matches the KA by requiring knowledge of the power supplys to the LPSW pumps.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT39 Q64

Development References

ILT39 Q64
SSS LPW Pg 31

Student References Provided

SYS075 K2.03 - Circulating Water System

Knowledge of bus power supplies to the following: (CFR: 41.7)

Emergency/essential SWS pumps ....

401-9 Comments:

Remarks/Status

SYS041 K1.02 - Steam Dump System (SDS)/Turbine Bypass Control
Knowledge of the Physical connections and/or cause-effect relationships between the SDS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
S/G level ....

Given the following Unit 1 conditions:

- Reactor power = 80%
- 1A TBVs (1MS-22 and 1MS-19) fail open

When the plant stabilizes from the event, the 1A SG level will be __(1)__ the pre-transient level and the plant MWe output will be __(2)__ the initial output .

Which ONE of the following completes the statement above?

A. 1. the same as
   2. the same as

B. 1. the same as
   2. lower than

C. 1. higher than
   2. the same as

D. 1. higher than
   2. lower than

General Discussion

Answer A Discussion

1st part is correct. When the TBVs fail open, SG pressure decreases. The reduction in pressure will also momentarily cause feedwater flow to increase, also causing level to initially increase. However as ICS stabilizes the plant, FDW flow returns to the pre transient value and Tave is restored to setpoint which results in SG level returning to pre-transient level.

2nd part is incorrect because when the event stabilizes, reactor power will be the same and some of the steam flow that was going to the turbine is now bypassing the turbine (less Mwe). It is plausible because reactor will be the same as the pre-transient power level.

Answer B Discussion

1st part is correct. When the TBVs fail open, SG pressure decreases. The reduction in pressure will also momentarily cause feedwater flow to increase, also causing level to initially increase. However as ICS stabilizes the plant, FDW flow returns to the pre transient value and Tave is restored to setpoint which results in SG level returning to pre-transient level..

2nd part is correct. When the plant stabilizes, the total steam demand will be ~ same however, some steam has been diverted from the turbine directly into the condenser. The lower amount of steam flowing through the turbine will reduce electrical output.

Answer C Discussion

1st part is incorrect. Plausible since SG levels do initially increase. When the TBVs fail open, SG pressure decreases. The reduction in pressure will also momentarily cause feedwater flow to increase, also causing level to initially increase. However as ICS stabilizes the plant, FDW flow returns to the pre transient value and Tave is restored to setpoint which results in SG level returning to pre-transient level.

2nd part is incorrect because when the event stabilizes, reactor power will be the same and some of the steam flow that was going to the turbine is now bypassing the turbine (less Mwe). It is plausible because reactor will be the same as the pre-transient power level.

Answer D Discussion

1st part is incorrect. Plausible since SG levels do initially increase. When the TBVs fail open, SG pressure decreases. The reduction in pressure will also momentarily cause feedwater flow to increase, also causing level to initially increase. However as ICS stabilizes the plant, FDW flow returns to the pre transient value and Tave is restored to setpoint which results in SG level returning to pre-transient level.

2nd part is correct. When the plant stabilizes, the total steam demand will be ~ same however, some steam has been diverted from the turbine directly into the condenser. The lower amount of steam flowing through the turbine will reduce electrical output.

Basis for meeting the KA

This question matches the KA by requiring knowledge of how a Turbine Bypass Valve malfunction will effect SG levels.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References
MS Dwg SAE-L O57 ICS Dwg

Student References Provided

SYS041 K1.02 - Steam Dump System (SDS)/Turbine Bypass Control

Knowledge of the Physical connections and/or cause-effect relationships between the SDS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

S/G level ....

401-9 Comments:	Remarks/Status

ILT 47 ONS SRO NRC Examination QUESTION 60 60 **C**

SYS055 K3.01 - Condenser Air Removal System (CARS)
Knowledge of the effect that a loss or malfunction of the CARS will have on the following: (CFR: 41.7 / 45.6)
Main condenser ....

Given the following Unit 1 conditions:

- Reactor power = 100%
- The operating CSAE malfunctions
- Condenser vacuum = 24.5" slowly decreasing
- AP/27 (Loss Of Condenser Vacuum) has been initiated
- Vacuum Pumps have been started

In accordance with AP/27, which ONE of the following states:

1) the MINIMUM vacuum that the Main Vacuum Pump must be pulling prior to opening its inlet valves?

2) the consequences if the above criteria is violated?

A. 1. 20" Hg Vacuum
2. The loss of vacuum may worsen

B. 1. 20" Hg Vacuum
2. The vacuum pump seal may be lost resulting in damage to the pump

C. 1. 24" Hg Vacuum
2. The loss of vacuum may worsen

D. 1. 24" Hg Vacuum
2. The vacuum pump seal may be lost resulting in damage to the pump

---

General Discussion

Answer A Discussion

1st part is incorrect because AP/24 states that a minimum of 24" Hg must be established before opening the vacuum pump inlet valves. It is plausible because in the lesson it provides 20" Hg as the vacuum that should be established by the vacuum pumps after running for 30 minutes on a startup.

2nd part is correct. This is caution stated in AP/24 prior to step 4 in Encl 5.1.

Answer B Discussion

1st part is incorrect because AP/24 states that a minimum of 24" Hg must be established before opening the vacuum pump inlet valves. It is plausible because in the lesson it provides 20" Hg as the vacuum that should be established by the vacuum pumps after running for 30 minutes on a startup.

2nd part is incorrect because the caution statement states that the loss of vacuum may increase. It is plausible because the water seal around the pump impeller is what forms the volute and allows the pump to work. Changing the DP across this pump could alter that water seal.

Answer C Discussion

1st part is correct. AP/27 Enclosure 5.1 step 4: IAAT a Main Vacuum Pump is pulling > 24" Vacuum, open the associated inlet valve.

2nd part is correct. This is caution stated in AP/24 prior to step 4 in Encl 5.1.

Answer D Discussion

1st part is correct. AP/27 Enclosure 5.1 step 4: IAAT a Main Vacuum Pump is pulling > 24" Vacuum, open the associated inlet valve.

2nd part is incorrect because the caution statement states that the loss of vacuum may increase. It is plausible because the water seal around the pump impeller is what forms the volute and allows the pump to work. Changing the DP across this pump could alter that water seal.

Basis for meeting the KA

This question matches the KA by requiring knowledge of how a malfunction / incorrect operation of the CARS (being aligned prior to sufficient vacuum being established) can have on condenser vacuum.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References
STG-CVS AP 27

Student References Provided

SYS055 K3.01 - Condenser Air Removal System (CARS)

Knowledge of the effect that a loss or malfunction of the CARS will have on the following: (CFR: 41.7 / 45.6)

Main condenser ....

401-9 Comments:

Remarks/Status

SYS056 A2.04 - Condensate System

Ability to (a) predict the impacts of the following malfunctions or operations on the Condensate System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Loss of condensate pumps ....

Given the following Unit 1 conditions:

Time = 1200:00

- Reactor power = 80% stable
- 1A and 1B CBP operating

Time = 1201:00

- 1A CBP trips
- Feedwater Pump suction pressure = 225 psig slowly decreasing

Time = 1203:00

- Feedwater Pump suction pressure = 220 slowly increasing

Which ONE of the following describes the:

1) runback rate (%/min) inserted at Time = 1201:00 to ICS?

2) procedure that will be directed by the CRS at Time = 1203:00?

A. 1. 15
2. AP/1/A/1700/001 (Unit Runback)

B. 1. 15
2. EOP

C. 1. 20
2. AP/1/A/1700/001 (Unit Runback)

D. 1. 20
2. EOP

ILT 47 ONS SRO NRC Examination QUESTION 61

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D

General Discussion

Answer A Discussion

1st part is Incorrect because the rate would be 20% / min. It is plausible since there are ICS runbacks that incorporate the 15%/min runback rate.

2nd part is incorrect because after 90 seconds, if FDWP suction pressure is still < 235 psig the FDWP's will trip which will trip the Rx and require entry into the EOP to mitigate the loss of main feedwater. It is plausible because it suction pressure returned before 90 seconds, it would be correct.

Answer B Discussion

1st part is Incorrect because the rate would be 20% / min. It is plausible since there are ICS runbacks that incorporate the 15%/min runback rate.

2nd part is correct. After 90 seconds, if FDWP suction pressure is still < 235 psig the FDWP's will trip which will trip the Rx and require entry into the EOP to mitigate the loss of main feedwater.

Answer C Discussion

1st part is correct. With FDWP suction pressure < 235 psig, an ICS runback is initiated. The runback rate is 20%/min to a power level of 15% or until the low suction pressure clears.

2nd part is incorrect because after 90 seconds, if FDWP suction pressure is still < 235 psig the FDWP's will trip which will trip the Rx and require entry into the EOP to mitigate the loss of main feedwater. It is plausible because it suction pressure returned before 90 seconds, it would be correct.

Answer D Discussion

Correct. With FDWP suction pressure < 235 psig, an ICS runback is initiated. The runback rate is 20%/min to a power level of 15% or until the low suction pressure clears. After 90 seconds, if FDWP suction pressure is still < 235 psig the FDWP's will trip which will trip the Rx and require entry into the EOP to mitigate the loss of main feedwater.

Basis for meeting the KA

Requires knowledge of the impact of a loss of Condensate Booster Pump and knowledge of the procedure that will be used to mitigate the event.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT44 (Q 49) NRC Exam

Development References

STG-ICS Ch 2 Pg 29
CF-FDW Pg 10
ILT44 Q49

Student References Provided

SYS056 A2.04 - Condensate System

Ability to (a) predict the impacts of the following malfunctions or operations on the Condensate System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Loss of condensate pumps ....

401-9 Comments:

Remarks/Status

SYS071 A3.03 - Waste Gas Disposal System (WGDS)
Ability to monitor automatic operation of the Waste Gas Disposal System including: (CFR: 41.7 / 45.5)
Radiation monitoring system alarm and actuating signals ....

Unit 1 plant conditions:

- A gaseous waste release at 1/3 station limit is being performed

1) The Alert and High setpoints for ____(1)____ are based on this limit.

2) If the High alarm setpoint is reached on ____(2)____, the gaseous waste release will be automatically terminated.

Which ONE of the following completes the statements above?

A. 1. 1RIA-38
   2. 1RIA-38

B. 1. 1RIA-38
   2. 1RIA-45

C. 1. 1RIA-45
   2. 1RIA-38

D. 1. 1RIA-45
   2. 1RIA-45

ILT 47 ONS SRO NRC Examination QUESTION 62

62

C

General Discussion

Answer A Discussion

1st part is incorrect because 1RIA-37 & 38 will be set "above background" based on RP sample reults. The setpoint for 1RIA-45 is specifically based on 1/3 station limit. It is plausible because 1-RIA-38 setpoints are adjusted prior to the release.

2nd part is correct. 1RIA-37 or 38 alarm will teminate the release.

Answer B Discussion

1st part is incorrect because 1RIA-37 & 38 will be set "above background" based on RP sample results. The setpoint for 1RIA-45 is specifically based on 1/3 station limit. It is plausible because 1-RIA-38 setpoints are adjusted prior to the release.

2nd part is incorrect because 1RIA-45 reaching its alarm setpoint will NOT isolate the gaseous waste release. It is plausible because it will isolate the RB Purge system if it is in operation.

Answer C Discussion

1st part is correct. When performing a gaseous waste release, RIA-45 setpoint is based on 1/3 station limit.

2nd part is correct. 1RIA-37 or 38 alarm will teminate the release.

Answer D Discussion

1st part is correct. When performing a gaseous waste release, RIA-45 setpoint is based on 1/3 station limit.

2nd part is incorrect because 1RIA-45 reaching its alarm setpoint will NOT isolate the gaseous waste release. It is plausible because it will isolate the RB Purge system if it is in operation.

Basis for meeting the KA

Requires knowledge of cause/effect relationship between the PRM System and the WGD system during releases

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	2009 Q64 NRC Exam

Development References
WE-GWD Pg 25 RAD RIA Pg 21, 23

Student References Provided

SYS071 A3.03 - Waste Gas Disposal System (WGDS)

Ability to monitor automatic operation of the Waste Gas Disposal System including: (CFR: 41.7 / 45.5)

Radiation monitoring system alarm and actuating signals ....

401-9 Comments:

Remarks/Status

SYS072 A4.03 - Area Radiation Monitoring (ARM) System
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
Check source for operability demonstration ....

Given the following Unit 1 conditions:
Initial conditions:

- Enclosure 4.9 (GWD Tank Release) of OP/1-2/A/1104/018 (GWD System) in progress

Current conditions:

- 1RIA-37 source check is to be performed

1) The source check __ (1) __ performed by actuating 1RIA-37 Source Check on the "Enable Controls" screen.

2) The source check is operable if __ (2) __.

Which ONE of the following completes the statements above?

A. 1. is
2. the Process Monitor Fault Alarm is NOT received

B. 1. is
2. 1RIA-37 readings increase during the source check

C. 1. is NOT
2. the Process Monitor Fault Alarm is NOT received

D. 1. is NOT
2. 1RIA-37 reading increase during the source check

ILT 47 ONS SRO NRC Examination QUESTION 63 63 **A**

General Discussion

Answer A Discussion

First part is correct. The source check is performed by actuating 1RIA-37 source Check on the "Enable Controls" screen.

Second part is correct. The source check is operable if Process Monitor Fault is NOT received.

Answer B Discussion

First part is correct. The source check is performed by actuating 1RIA-37 source Check on the "Enable Controls" screen.

Second part incorrect because 1RIA-37 readings should not go up during a source check. It is plausible because it is a common misconception that the RIA readings will increase on a source check.

Answer C Discussion

First part is incorrect because it is the correct method. It is plausible because this answer would be correct for 1RIA-38.

Second part is correct. The source check is operable if Process Monitor Fault is NOT received.

Answer D Discussion

First part is incorrect because it is the correct method. It is plausible because this answer would be correct for 1RIA-38.

Second part is plausible because it is a common misconception that the RIA readings will increase on a source check.

Basis for meeting the KA

Question requires knowledge of how a source check is performed and the expected RIA response.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT42 Q52 NRC Exam

Development References
ILT42 Q52 RAD RIA Pg 36

Student References Provided

SYS072 A4.03 - Area Radiation Monitoring (ARM) System

Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)

Check source for operability demonstration ....

401-9 Comments:	Remarks/Status

ILT 47 ONS SRO NRC Examination QUESTION 64 64 **B**

SYS079 K4.01 - Station Air System (SAS)
Knowledge of SAS design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)
Cross-connect with IAS ....

Given the following conditions:

- Time = 0400
- IA header pressure = 88 psig decreasing

At 0400 the Diesel Air Compressors are __ (1) __ and SA-141 (SA to IA Controller) is __ (2) __.

Which ONE of the following completes the statement above?

A. 1. Operating
2. Open

B. 1. Operating
2. Closed

C. 1. Shutdown
2. Open

D. 1. Shutdown
2. Closed

B

General Discussion

Answer A Discussion

1st part is correct. DG air compressors start if IA pressure decreases to 90 psig.

2nd part is incorrect because SA-141 does not open until IA pressure decreases to 85 psig. It is plausible because there are numerous setpoint between 80 and 100 psig. (Compressor setpoints of 100, 95, 93, 90 psi) If pressure were a few psi lower, it would be correct.

Answer B Discussion

1st part is correct. DG air compressors start if IA pressure decreases to 90 psig.

2nd part is correct. SA-141 does not open until IA pressure decreases to 85 psig.

Answer C Discussion

1st part is incorrect because the DG air compressor would be operating. It is plausible because there are numerous setpoint between 80 and 100 psig. It pressure were a few psi higher, it would be correct.

2nd part is incorrect because SA-141 does not open until IA pressure decreases to 85 psig. It is plausible because there are numerous setpoint between 80 and 100 psig. (Compressor setpoints of 100, 95, 93, 90 psi) It pressure were a few psi lower, it would be correct.

Answer D Discussion

1st part is incorrect because the DG air compressor would be operating. It is plausible because there are numerous setpoint between 80 and 100 psig. It pressure were a few psi higher, it would be correct.

2nd part is correct. SA-141 does not open until IA pressure decreases to 85 psig.

Basis for meeting the KA

Requires knowledge of automatic cross-connect between Service air and
Instrument air systems.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT41 Q65 NRC Exam

Development References

SSS-IA Pg 42, 45
2010A Q64

Student References Provided

SYS079 K4.01 - Station Air System (SAS)
Knowledge of SAS design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)
Cross-connect with IAS ....

401-9 Comments:

Remarks/Status

SYS015 A1.08 - Nuclear Instrumentation System (NIS)

Ability to predict and/or monitor changes in parameters to prevent exceeding design limits) associated with operating the NIS controls including: (CFR: 41.5 . 45.5)

Changes in RCS temperature ....

Given the following Unit 1 conditions:
Initial Conditions:

- Time = 1200
- Power escalation in progress
- Core Thermal Power = 50% slowly increasing
- NI Power = 52% slowly increasing

Current Conditions:

- Time = 1400
- Core Thermal Power = 60% slowly increasing

1) At Time = 1200 NI's are considered __(1)__.

2) As a result of changes in RCS temperature, at Time = 1400 NI's will be __(2)__ than 2% different than Core Thermal Power.

Which ONE of the following completes the statements above?

A. 1. conservative
2. less

B. 1. conservative
2. greater

C. 1. non-conservative
2. less

D. 1. non-conservative
2. greater

ILT 47 ONS SRO NRC Examination QUESTION 65

65

A

General Discussion

Answer A Discussion

First part is correct. NI's are conservative if they read > core thermal power.

Second part is correct since T cold is the primary driver of RCS leakage out of the core and as power increases from 50% to 60% Tcold decrease which results in a lower percentage of neutrons leaking out of the core which means relative to core thermal power, NI indication would be less

Answer B Discussion

First part is correct. NI's are conservative if they read > core thermal power.

Second part is incorrect but plausible since Thot increase as power increases and therefore it would be easy to assume Thot is the primary driver of NI leakage out of the core which would lead to choosing this answer.

Answer C Discussion

First part is incorrect but plausible since it is an easily confused issue as to when NI's are conservative vs non-conservative. It would be easy to believe that if NI's indicate higher then they would be non-conservative

Second part is correct since T cold is the primary driver of RCS leakage out of the core and as power increases from 50% to 60% Tcold decrease which results in a lower percentage of neutrons leaking out of the core which means relative to core thermal power, NI indication would be less

Answer D Discussion

First part is incorrect but plausible since it is an easily confused issue as to when NI's are conservative vs non-conservative. It would be easy to believe that if NI's indicate higher then they would be non-conservative

Second part is incorrect but plausible since Thot increase as power increases and therefore it would be easy to assume Thot is the primary driver of NI leakage out of the core which would lead to choosing this answer.

Basis for meeting the KA

Requires ability to predict changes in NI indication based on changes in RCS temperature.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References
IC-NI

Student References Provided

SYS015 A1.08 - Nuclear Instrumentation System (NIS)

Ability to predict and/or monitor changes in parameters to prevent exceeding design limits) associated with operating the NIS controls including: (CFR: 41.5 . 45.5)

Changes in RCS temperature ....

401-9 Comments:

Remarks/Status

GEN2.1 2.1.15 - GENERIC - Conduct of Operations

Conduct of Operations

Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, Operations memos, etc.

Given the following Unit 1 plant conditions:

- Reactor Power = 100%
- An evolution is to be conducted during the shift
- A related annunciator alarm has been deemed “expected” during the pre-job brief

0800:

- The expected alarm is received

0805:

- The expected alarm clears

In accordance with AD-OP-ALL-1000 (Conduct of Operations)…

1) Shift Manager permission __(1)__ required to suspend the requirement to verbally announce the alarm.

2) At 0805, the operator ____(2)____ required to report to the CRS that the alarm cleared.

Which ONE of the following completes the statements above?

A. 1. is
2. is

B. 1. is
2. is NOT

C. 1. is NOT
2. is

D. 1. is NOT
2. is NOT

ILT 47 ONS SRO NRC Examination QUESTION 66

66

D

General Discussion

Answer A Discussion

1st part is incorrect because the CRS can suspend the requirement to verbally announce alarms. Assuming that this is part of the prcedure in use and or part of the pre-job brief. It is plausible because the Shift Manager is responsible for ensuring that plant operations are conducted in accordance with required procedures (including normal alarm protocol).

2nd part is incorrect because for "expected" alarms, when they clear, it is not required to be reported. It is plausible because for expected alarms, it would be correct.

Answer B Discussion

1st part is incorrect because the CRS can suspend the requirement to verbally announce alarms. Assuming that this is part of the prcedure in use and or part of the pre-job brief. It is plausible because the Shift Manager is responsible for ensuring that plant operations are conducted in accordance with required procedures (including normal alarm protocol)..

2nd part is correct. For expected alarms, when they clear, Per AD-OP-ALL-1000 they are not required to be reported.

Answer C Discussion

1st part iscorrect. The Shift Manager's permission is not required to suspend announcing of annunciators.

2nd part is incorrect because for "expected" alarms, when they clear, it is not required to be reported. It is plausible because for expected alarms, it would be correct.

Answer D Discussion

1st part iscorrect. The Shift Manager's permission is not required to suspend announcing of annunciators.

2nd part is correct. For expected alarms, when they clear, Per AD-OP-ALL-1000 they are not required to be reported.

Basis for meeting the KA

Question matches the KA by requiring knowledge of the adminestrative requirements for temporary procedure use (in this case, the abstaining of normal procedure adherence).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References
AD-OP-ALL-1000 ADM-OMP, Obj: R58

Student References Provided

GEN2.1 2.1.15 - GENERIC - Conduct of Operations

Conduct of Operations

Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, Operations memos, etc.

401-9 Comments:

Remarks/Status

GEN2.1 2.1.25 - GENERIC - Conduct of Operations
Conduct of Operations
Ability to interpret reference materials, such as graphs, curves, tables, etc. (CFR: 41.10 / 43.5 / 45.12)

Given the following Unit 3 conditions:
- Time = 1200
- LDST level = 75 inches decreasing
- LDST pressure = 35 psig slowly decreasing

Which ONE of the following describes the:

1) status of the HPI system at Time = 1200?

2) required action in accordance with OP/1108/001 (Curves and General Information)?

REFERENCE PROVIDED

A. 1. Operable
2. Initiate makeup to LDST

B. 1. Operable
2. Depressurize LDST

C. 1. Inoperable
2. Initiate makeup to LDST

D. 1. Inoperable
2. Depressurize LDST

General Discussion

Answer A Discussion

First part is incorrect. Operation above and to the left of curve 1 requires
declaring both trains of HPI inoperable. Plausible because the candidate must recall
from memory the actions based on the location of level/press on the curve.

Second part is incorrect. Making up would increase level which would also increase pressure and keep the LDST out of the permissible region. Plausible because the
compensatory actions may be applied for Step 1 “HPI Pumps Operating” in that the
student may select this section based on having HPI in service. Could also be
chosen if the candidate misapplies the required actions for being outside the
“Permissible Operating Region” but still between curve 1 and curve 2.

Answer B Discussion

First part is incorrect. Operation above and to the left of curve 1 requires
declaring both trains of HPI inoperable. Plausible because the candidate must recall
from memory the actions based on the location of level/press on the curve.

Second part is correct. Depressurizing the LDST would bring the LDST back into the permissible operation region.

Answer C Discussion

First part is correct. The current operating point is above and to the left of curve 1 which makes it inoperable.

Second part is incorrect. Making up would increase level which would also increase pressure and keep the LDST out of the permissible region. Plausible because the
compensatory actions may be applied for Step 1 “HPI Pumps Operating” in that the
student may select this section based on having HPI in service. Could also be
chosen if the candidate misapplies the required actions for being outside the
“Permissible Operating Region” but still between curve 1 and curve 2.

Answer D Discussion

First part is correct. The current operating point is above and to the left of curve 1 which makes it inoperable.

Second part is correct. Depressurizing the LDST would bring the LDST back into the permissible operation region.

Basis for meeting the KA

Requires the ability to diagnose trends, to apply the trend to reference material, and
knowledge of corrective actions

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	2009 (Q 75) NRC Exam

Development References

PNS-HPI Pg 39
OP/0/A/1108/001 Enclosure 4.39
2009 Q75

Student References Provided

OP/0/A/1108/001
Enclosure 4.39 (page 1 only with action notes removed)

GEN2.1 2.1.25 - GENERIC - Conduct of Operations
Conduct of Operations
Ability to interpret reference materials, such as graphs, curves, tables, etc. (CFR: 41.10 / 43.5 / 45.12)

401-9 Comments:	Remarks/Status

GEN2.1 2.1.28 - GENERIC - Conduct of Operations
Conduct of Operations
Knowledge of the purpose and function of major system components and controls. (CFR: 41.7)

Given the following Unit 1 conditions:

- RCS pressure = 525 psig stable
- An attempt is made to open 1LP-1 (LPI RETURN BLOCK FROM RCS)

1) 1LP-1 __ (1) __ open.

2) The reason 1LP-1 has an interlock is to __ (2) __.

Which ONE of the following completes the statements above?

A. 1. will
   2. prevent over pressurizing LPI suction piping

B. 1. will
   2. ensure delta p across 1LP-1 will allow it to open

C. 1. will NOT
   2. prevent over pressurizing LPI suction piping

D. 1. will NOT
   2. ensure delta p across 1LP-1 will allow it to open

ILT 47 ONS SRO NRC Examination QUESTION 68

68

C

General Discussion

Answer A Discussion

First part is incorrect and plausible. The 1 LP-1 interlock prevents 1LP-1 from being opened when WR RCS pressure (via the Amphenol connector) is >400 psig. At 550 psig ES would normally have actuated the LPI system on a low RCS pressure. It may be incorrectly assumed that since LPI actuates at 550 psi that it must be OK to open 1LP-1.

Second part is correct. The interlock is designed to prevent over pressurizing LPI suction piping.

Answer B Discussion

First part is incorrect and plausible. The 1 LP-1 interlock prevents 1LP-1 from being opened when WR RCS pressure (via the Amphenol connector) is >400 psig. At 550 psig ES would normally have actuated the LPI system on a low RCS pressure. It may be incorrectly assumed that since LPI actuates at 550 psi that it must be OK to open 1LP-1.

Second part is incorrect and plausible. Waiting on a lower RCS pressure to open 1LP-1 would in fact lower the dp across 1LP-1 when it is opened. There are many different valves througho

Answer C Discussion

First part is correct. The 1LP-1 interlock prevents 1LP-1 from being opened when WR RCS pressure (via the Amphenol connector) is >400 psig.

Second part is correct. The interlock is designed to prevent over pressurizing LPI suction piping.

Answer D Discussion

First part is correct. The 1LP-1 interlock prevents 1LP-1 from being opened when WR RCS pressure (via the Amphenol connector) is >400 psig.

Second part is incorrect and plausible. Waiting on a lower RCS pressure to open 1LP-1 would in fact lower the dp across 1LP-1 when it is opened. There are many different valves throughout the plant where we take specific actions to ensure dp is low enough across a valve before we try to open it (Ex. MSCV's, FDW valves, etc.).

Basis for meeting the KA

Requires knowledge of how LPI suction piping overpressure protection is accomplished. This is done by an interlock that prevents placing LPI DHR piping in service prior to being below 400 psi,

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT39 (Q 31) NRC Exam

Development References
PNS-LPI Pg 49, 52 ILT39 Q31

Student References Provided

GEN2.1 2.1.28 - GENERIC - Conduct of Operations

Conduct of Operations

Knowledge of the purpose and function of major system components and controls. (CFR: 41.7)

401-9 Comments:

Remarks/Status

GEN2.2 2.2.3 - GENERIC - Equipment Control

Equipment Control

(multi-unit license) Knowledge of the design, procedural, and operational differences between units. (CFR: 41.5 / 41.6 / 41.7 / 41.10 / 45.12)

Given the following Unit 3 conditions:

- Reactor power = 100%
- 3RC-1 has failed OPEN
- 3RC-3 will NOT close
- RCS pressure continues to decrease

Which ONE of the following describes the Reactor Coolant Pump(s) that will be INITIALLY secured after the Reactor has been Manually tripped in accordance with AP/3/A/1700/044 (Abnormal Pressurizer Pressure Control)?

A. 3B1 ONLY

B. 3B1 AND 3B2

C. 3A1 ONLY

D. 3A1 AND 3A2

ILT 47 ONS SRO NRC Examination QUESTION 69 69 **B**

General Discussion

Answer A Discussion

Incorrect: Plausible since on Unit 3 the Pzr spray line is located on the discharge of the 3B1 RCP and therefore securing this pump alone would significantly decrease the amount of Pzr spray through the failed open valves. Since the question asks which pumps will be "initially" secured it is plausible to believe that the AP would direct securing the spray pump only and then only securing other pumps if this were not sufficient. Additionally plausible due to the process used to choose what pump to leave running and why. It is common practice to always leave the spray pump running when possible. In that context, leaving both pumps on in the loop with Pzr spray is not considered (as a function of ensuring Pzr spray available) therefore it would be plausible to believe that you only need to secure the RCP in the loop with the Pzr spray tap.

Answer B Discussion

Correct: AP/44 directs tripping the Rx and securing both the 3B1 and the 3B2 RCP's if RCS pressure cannot be controlled using 3RC-1 and 3RC-3.

Answer C Discussion

Incorrect: Plausible since the Pzr spray line is located on the discharge of the 1A1 RCP on unit 1 therefore securing the 3A1 RCP only is plausible based on the misconception that the Pzr spray line is on the A loop on Unit 3 as well. Under that misconception, securing the 3A1 RCP would significantly reduce spray flow through the failed valves and therefore make this choice plausible. since the question asks which pumps will be "initially" secured it is plausible to believe that the AP would direct securing the spray pump only and then only securing other pumps if this were not sufficient. Additionally plausible due to the process used to choose what pump to leave running and why. It is common practice to always leave the spray pump running when possible. In that context, leaving both pumps on in the loop with Pzr spray is not considered (as a function of ensuring Pzr spray available) therefore it would be plausible.

Answer D Discussion

Incorrect: Plausible since this would be correct if the event occurred on Unit 1.

Basis for meeting the KA

Knowing the difference in Unit 1 vs. Unit 2&3 with regards to location of the Pzr spray line and differences in direction provided in AP/44 in relation to failed open spray valve and associated block valve meet the KA.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT40 (Q 7) NRC Exam

Development References

EAP-APG
3 AP/44
ILT40 Q7

Student References Provided

GEN2.2 2.2.3 - GENERIC - Equipment Control
Equipment Control
(multi-unit license) Knowledge of the design, procedural, and operational differences between units. (CFR: 41.5 / 41.6 / 41.7 / 41.10 / 45.12)

401-9 Comments:	Remarks/Status

GEN2.2 2.2.17 - GENERIC - Equipment Control

Equipment Control

Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator. (CFR: 41.10 / 43.5 / 45.13)

Given the following Unit 1 conditions:

- Reactor power is being reduced from 100% to 88% in order to perform surveillance testing
- OP/1/A/1102/004 (Operation at Power), Enclosure 4.2 (Power Reduction) is in progress

1) The SOC ____(1)____ required to be notified.

2) A Maneuvering Plan ____(2)____ required to be generated.

Which ONE of the following completes the above statements for the power reduction?

A. 1. is
   2. is

B. 1. is
   2. is NOT

C. 1. is NOT
   2. is

D. 1. is NOT
   2. is NOT

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A

General Discussion

Answer A Discussion

1st part is correct. When using OP/1/A/1102/004, you are required to notify the SOC anytime that power is reduced.

2nd part is correct because a manuvering plan is required anytime that you are reducing power more than 10%.

Answer B Discussion

1st part is correct. When using OP/1/A/1102/004, you are required to notify the SOC anytime that power is reduced.

2nd part is incorrect because a maneuvering plan is required. It is plausible because some notification requirements are not required unless a 15% power change occurs (Primary Chemistry).

Answer C Discussion

1st part is incorrect because the SOC is required to be notified if a planned 12% power reduction is to occur. It is plausible because there are several thresholds for notifications during a power reduction (2nd chem @ 6%, primary chemistry @ 15%, maneuvering plan @ 10%) . There is not a threshold for notifying the SOC however.

2nd part is correct because a manuvering plan is required anytime that you are reducing power more than 10%.

Answer D Discussion

1st part is incorrect because the SOC is required to be notified if a 12% planned power reduction is to occur. It is plausible because there are several thresholds for notifications during a power reduction (2nd chem @ 6%, primary chemistry @ 15%, maneuvering plan @ 10%) . There is not a threshold for notifying the SOC however.

2nd part is incorrect because a maneuvering plan is required. It is plausible because some notification requirements are not required unless a 15% power change occurs (Primary Chemistry).

Basis for meeting the KA

This question matches the KA by requiring knowledge of process of coordinating with the transmission system operator (SOC) during maintenance activities.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

OP 1 A 1102 004 Encl 4.2
CP 12

Student References Provided

GEN2.2 2.2.17 - GENERIC - Equipment Control

Equipment Control

Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator. (CFR: 41.10 / 43.5 / 45.13)

401-9 Comments:

Remarks/Status

GEN2.3 2.3.7 - GENERIC - Radiation Control
Radiation Control
Ability to comply with radiation work permit requirements during normal orabnormal conditions. (CFR: 41.12 / 45.10)

Given the following Unit 1 conditions:

- An AO is to valve out the 1A Seal Supply filter
- Anticipated dose rate alarms were briefed by RP

1) Based on the RWP and the Plan View, the maximum time below that can be taken to perform this task per PD-RP-ALL-0001, Radiation Worker Responsibilities before the AO is expected to exit the area is __ (1) __ minutes.

2) Upon receipt of a second dose rate alarm that was anticipated and previously briefed, the AO __ (2) __.

Which ONE of the following completes the statement above?

REFERENCE PROVIDED

A. 1. 21
2. may continue to work

B. 1. 21
2. must immediately exit the area

C. 1. 27
2. may continue to work

D. 1. 27
2. must immediately exit the area

General Discussion

Answer A Discussion

1st part is correct. Per PD-RP-ALL-0001 (Radiation Worker Responsibilities), rad workers are expected to exit the work area when the ED accumulates 80% of the dose alarm setpoint.

2nd part is correct. Per NSD 507 the NEO can continue to work if a peviously briefed dose rate alarm occurs. One the third alarm he must leave the area.

Answer B Discussion

1st part is correct. Per PD-RP-ALL-0001 (Radiation Worker Responsibilities), rad workers are expected to exit the work area when the ED accumulates 80% of the dose alarm setpoint.

2nd part is incorrect. It is plausible because if it were the 3rd alarm or an unexpected alarm, it would be correct.

Answer C Discussion

1st part is incorrect because its based on the Dose alarm setopint when per PD-RP-ALL-0001 (Radiation Worker Responsibilities), rad workers are expected to exit the work area when the ED accumulates 80% of the dose alarm setpoint. It is plausible because it is the correct calculation if you were allowed to stay until your dose alarm sounded.

2nd part is correct. Per NSD 507 the NEO can continue to work if a peviously briefed dose rate alarm occurs. One the third alarm he must leave the area.

Answer D Discussion

1st part is incorrect because its based on the Dose alarm setopint when per PD-RP-ALL-0001 (Radiation Worker Responsibilities), rad workers are expected to exit the work area when the ED accumulates 80% of the dose alarm setpoint. It is plausible because it is the correct calculation if you were allowed to stay until your dose alarm sounded.

2nd part is incorrect. It is plausible because if it were the 3rd alarm or an unexpected alarm, it would be correct.

Basis for meeting the KA

Requires knowledge of how to use radiation work permits and RWPs to determine stay time.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	ILT41 Q73

Development References

RAD-RPP Pg 48
ILT41 Q73
Plan View SS Filter Room
RWP
PD RP ALL 0001 Pg 9

Student References Provided

Plan View
RWP 23

GEN2.3 2.3.7 - GENERIC - Radiation Control
Radiation Control
Ability to comply with radiation work permit requirements during normal orabnormal conditions. (CFR: 41.12 / 45.10)

401-9 Comments:

Remarks/Status

GEN2.3 2.3.11 - GENERIC - Radiation Control
Radiation Control
Ability to control radiation releases. (CFR: 41.11 / 43.4 / 45.10)

Given the following Plant conditions:

- Spent Fuel Storage Cask has been dropped in Unit 1&2 SFP
- Spent Fuel damage is visible
- RIA-6 and RIA-41 HIGH alarm actuates
- Spent Fuel Pool level = -3.5 feet decreasing

Which ONE of the following describes the

1) RB Purge filters that will be used to reduce off site releases

2) status of any SF Pumps that were in operation at the time of the event?

A. 1. Unit 1
2. ON

B. 1. Unit 1
2. OFF

C. 1. Unit 2
2. ON

D. 1. Unit 2
2. OFF

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General Discussion

Answer A Discussion
Incorrect: First part is Plausible since Unit 1 and Unit 2 share Spent Fuel Pools and there is only one set of filters needed for the Spent Fuel Filtered Exhaust system.

Since there are no dedicated filters, Unit 2's filters Reactor Building Purge filters are used. Second part is plausible since 4' is the level at which SF Pumps loose suction and level is still > 4 feet.

Answer B Discussion
Incorrect: First part is Plausible since Unit 1 and Unit 2 share Spent Fuel Pools and there is only one set of filters needed for the Spent Fuel Filtered Exhaust system.

Since there are no dedicated filters, Unit 2's filters Reactor Building Purge filters are used. Second part is correct

Answer C Discussion
First part is correct. Unit 1 and Unit 2 share Spent Fuel Pools and there is only one set of filters needed for the Spent Fuel Filtered Exhaust system. Unit 2 is used.

Second part is incorrect because the pumps will trip off at -2.5 feet. It is plausible since 4' is the level at which SF Pumps loose suction and level is still > 4 feet.

Answer D Discussion
First part is correct. Unit 1 and Unit 2 share Spent Fuel Pools and there is only one set of filters needed for the Spent Fuel Filtered Exhaust system. Unit 2 is used.

Second part is correct. The Spent Fuel Cooling pumps have a low level trip at -2.5 feet. Since level is -3.5 feet the pumps would be off.

Basis for meeting the KA
Question matches the KA by requiring knowledge of how radiation releases are controlled when conditions are abnormal.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	2010A (Q 59) NRC Exam

Development References
FH-SFC Pg 11
AP 9
2010A Q59

Student References Provided

GEN2.3 2.3.11 - GENERIC - Radiation Control
Radiation Control
Ability to control radiation releases. (CFR: 41.11 / 43.4 / 45.10)

401-9 Comments:

Remarks/Status

GEN2.4 2.4.1 - GENERIC - Emergency Procedures / Plan
Emergency Procedures / Plan
Knowledge of EOP entry conditions and immediate action steps. (CFR: 41.10 / 43.5 / 45.13)

Given the following Unit 1 conditions:
Initial conditions:

- Reactor power = 100%
- Both Main Feedwater pumps trip

Current conditions:

- REACTOR TRIP pushbutton has been depressed
- Reactor power = 4% slowly decreasing

Which ONE of the following describes the NEXT action required in accordance with EOP Immediate Manual Actions?

A. Perform Rule 1 (ATWS)

B. Manually insert control rods

C. Verify RCP seal injection available

D. Depress the Turbine TRIP pushbutton

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73

D

General Discussion

Answer A Discussion

Incorrect: Plausible since this would be correct if power level was > 5%.. Additional plausibility since there is a 1% power threshold for actions within Rule 2 therefore it is plausible to believe that if power is still > 1%, going to Rule 1 is required.

Answer B Discussion

Incorrect: Plausible since this is one of the first actions taken by Rule 1 during an ATWS. It is plausible to believe these actions are part of IMA's since it is in IMA's that the ATWS is diagnosed and inserting control rods is critical to the successful mitigation of the ATWS.

Answer C Discussion

Incorrect: Plausible since this is an action taken in IMA's however it is done after the main turbine is tripped.

Answer D Discussion

Correct: Since Rx power is < 5% the next action is to depress the Turbine Trip pushbutton.

Basis for meeting the KA

Requires the ability to perform EOP IMA's from memory.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT40 (Q 75) NRC Exam

Development References

EAP - IMAs & Symptom Check Pg 9
IMA's of EOP
Rule 1
ILT40 Q75

Student References Provided

GEN2.4 2.4.1 - GENERIC - Emergency Procedures / Plan

Emergency Procedures / Plan

Knowledge of EOP entry conditions and immediate action steps. (CFR: 41.10 / 43.5 / 45.13)

401-9 Comments:

Remarks/Status

ILT 47 ONS SRO NRC Examination QUESTION 74 74 **A**

GEN2.4 2.4.26 - GENERIC - Emergency Procedures / Plan

Emergency Procedures / Plan

Knowledge of facility protection requirements, including fire brigade andportable fire fighting equipment usage. (CFR: 41.10 / 43.5 / 45.12)

Given the following Unit 1 conditions:

- Reactor power = 100%
- 1SA3/B6 (FIRE ALARM) actuated
- Fire Alarm panel indication
    - point 0202071 (Unit 1 pipe trench room 348 North End) actuated
    - point 0202072 (Unit 1 pipe trench room 348 East Side) actuated

1) MERT will be dispatched to the area ____(1)____.

2) If the fire is determined to be in a cable tray, it ____(2)____ considered to be a "Challenging" fire.

Which ONE of the following completes the statements above?

A. 1. at the same time as the fire brigade
   2. is

B. 1. at the same time as the fire brigade
   2. is NOT

C. 1. ONLY after the fire is confirmed
   2. is

D. 1. ONLY after the fire is confirmed
   2. is NOT

General Discussion

Answer A Discussion

1st part is correct. Per 1SA3/B-6, IAAT two or more detectors are in alarm in the same zone: Dispatch Fire bridade and MERT.

2nd part iscorrect. A fire that is burning cables which have the potential to qaffect additional equipment is considered "challenging".

Answer B Discussion

1st part is correct. Per 1SA3/B-6, IAAT two or more detectors are in alarm in the same zone: Dispatch Fire bridade and MERT.

2nd part is incorrect because it is considered a challenging fire. It is plausible because the location (pipe trench room) doesn't seem as if it would be considered "challenging".

Answer C Discussion

1st part is incorrect because the MERT is send with the fire bridage upon receiving two alarms in the same area. It is plausible because if it were only one alarm in the same area, it would be correct.

2nd part is correct. A fire that is burning cables which have the potential to qaffect additional equipment is considered "challenging".

Answer D Discussion

1st part is incorrect because the MERT is send with the fire bridage upon receiving two alarms in the same area. It is plausible because if it were only one alarm in the same area, it would be correct.

2nd part is incorrect because it is considered a challenging fire. It is plausible because the location (pipe trench room) doesn't seem as if it would be considered "challenging".

Basis for meeting the KA

Question matches the KA by requiring knowledge of requirements for dispatching fire brigade / MERT when receiving a fire alarm.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References
IC-FDS Pg 11 ARG for 1SA3/B6 RP 1000 29

Student References Provided

GEN2.4 2.4.26 - GENERIC - Emergency Procedures / Plan

Emergency Procedures / Plan

Knowledge of facility protection requirements, including fire brigade andportable fire fighting equipment usage. (CFR: 41.10 / 43.5 / 45.12)

401-9 Comments:	Remarks/Status

GEN2.4 2.4.29 - GENERIC - Emergency Procedures / Plan
Emergency Procedures / Plan
Knowledge of the emergency plan. (CFR: 41.10 / 43.5 / 45.11)

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1) The on-site emergency facility that assumes responsibility for communications with offsite agencies including the NRC once it is activated is the __ (1) __.

2) The minimum level of emergency classification that always requires activation of the TSC and OSC is a(n) __ (2) __

Which ONE of the following completes the statements above?

A. 1. Technical Support Center (TSC)
2. Alert

B. 1. Technical Support Center (TSC)
2. Unusual Event

C. 1. Operations Support Center (OSC)
2. Alert

D. 1. Operations Support Center (OSC)
2. Unusual Event

General Discussion

Answer A Discussion

1st part is correct. The TSC assumes responsibility for off site communications once it is activated.

2nd part is correct. An Alert is the minimum classification that always requires activation of the TSC and OSC.

Answer B Discussion

1st part is correct. The TSC assumes responsibility for off site communications once it is activated.

Second part incorrect because the TSC does NOT have to be activated for an Unusual Event. It is plausible because the TSC and OSC can be activated in an Unusual Event but are not required to be.

Answer C Discussion

Firat part is incorrect because the TSC assumes responsibility for offsite communicaions. It is plausible because the OSC is responsible for onsite communications.

Second part is correct. An Alert is the minimum classification that always requires activation of the TSC and OSC.

Answer D Discussion

Firat part is incorrect because the TSC assumes responsibility for offsite communicaions. It is plausible because the OSC is responsible for onsite communications.

Second part incorrect because the TSC does NOT have to be activated for an Unusual Event. It is plausible because the TSC and OSC can be activated in an Unusual Event but are not required to be.

Basis for meeting the KA

This question matches the KA by requiring the applicant to have knowlwedge of the emergency plan.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT41 (Q 74) NRC Exam

Development References

EAP-SEP Pg 12, 15
ILT41 Q74

Student References Provided

GEN2.4 2.4.29 - GENERIC - Emergency Procedures / Plan
Emergency Procedures / Plan
Knowledge of the emergency plan. (CFR: 41.10 / 43.5 / 45.11)

401-9 Comments:

Remarks/Status