ILT48 ONS SRO NRC Examination **QUESTION 1** 1 **B**

EPE007 EK2.03 - Reactor Trip
Knowledge of the interrelations between a reactor trip and the following: (CFR 41.7 / 45.7)
Reactor trip status panel ....

Which ONE of the following combinations of statalarms, if actuated, would illuminate the Trip Confirm Lamp on the Diamond?

A. 1SA-1/E-2 (CRD BKR A TRIP)
1SA-1/E-4 (CRD BKR C TRIP)

B. 1SA-1/E-2 (CRD BKR A TRIP)
1SA-1/E-5 (CRD BKR D TRIP)

C. 1SA-1/E-2 (CRD BKR A TRIP)
1SA-1/E-6 (CRD ELECTRONIC TRIP E)

D. 1SA-1/E-3 (CRD BKR B TRIP)
1SA-1/E-7 (CRD ELECTRONIC TRIP F)

ILT48 ONS SRO NRC Examination QUESTION 1

1

B

General Discussion

Answer A Discussion

Incorrect. CRD Breaker trip combinations are: A and B, A and D, B and C, C and D, A and Electronic Trip F, B and Electronic Trip E. Plausible in that two CRD breakers indicate tripped and if either B or D were tripped in the place of C, it would be correct.

Answer B Discussion

CORRECT. CRD Breakers A and D tripped will de-energize the CRDs and initiate a Trip Confirm, which will cause the Trip Confirm Lamp on the Diamond to illuminate.

Answer C Discussion

Incorrect. CRD Breaker trip combinations are: A and B, A and D, B and C, C and D, A and Electronic Trip F, B and Electronic Trip E. Plausible since CRD BKR A and CRD ELECTRONIC TRIP F would be correct.

Answer D Discussion

Incorrect. CRD Breaker trip combinations are: A and B, A and D, B and C, C and D, A and Electronic Trip F, B and Electronic Trip E. Plausible since CRD BKR B and CRD ELECTRONIC TRIP E would be correct.

Basis for meeting the KA

The Reactor Trip Status at Oconee is indicated by the CRD Breaker statalarms on 1SA-1 and the Trip Confirm Lamp on the Diamond. The question requires knowledge of the interrelations between a reactor trip and the CRD Breaker statalarms and Trip Confirm Lamp on the Diamond.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

IC-CRI Obj. 05 and 14
ARG

Student References Provided

EPE007 EK2.03 - Reactor Trip
Knowledge of the interrelations between a reactor trip and the following: (CFR 41.7 / 45.7)
Reactor trip status panel ....

401-9 Comments:

Remarks/Status

APE008 AA2.24 - Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)
Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: (CFR: 43.5 / 45.13)
Value at which turbine bypass valve maintains header pressure after a reactor trip ....

Given the following Unit 1 conditions:

- Reactor power = 100%
- 1RC-66 fails OPEN

1) The initial Reactor Protective System trip function setpoint reached will be RCS __(1)__ pressure.

2) Following the Reactor trip, with actual SG Outlet pressure = 985 psig, the Turbine Bypass valves will be __(2)__.

Which ONE of the following completes the statements above?

ASSUME NO OPERATOR ACTIONS

A. 1. Low
2. throttled open

B. 1. Variable Low
2. throttled open

C. 1. Low
2. closed

D. 1. Variable Low
2. closed

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2

D

General Discussion

Answer A Discussion

Incorrect. First part is incorrect. Plausible since RCS pressure will decrease during this event to the point that the reactor will trip. Second part is incorrect. Plausible since it would be correct prior to the reactor trip. Normal Turbine Header Pressure setpoint = 885 psig. The Turbine Bypass Valves (TBVs) provide overpressure relief when the turbine is on-line at setpoint + 50 psig. So prior to the reactor trip, the TBVs would throttle open at 935 psig. Trip confirm inputs a + 125 psig bias. 885 + 125 = 1010 psig setpoint for TBVs following the reactor trip, so at 985 psig, the TBVs would be closed.

Answer B Discussion

Incorrect. First part is correct. 1RC-66 open at 100% is essentially a SBLOCA and will cause RCS pressure to decrease with little/no temperature decrease. This will cause a trip on Variable Low Pressure. Second part is incorrect. Plausible since it would be correct prior to the reactor trip. Normal Turbine Header Pressure setpoint = 885 psig. The Turbine Bypass Valves (TBVs) provide overpressure relief when the turbine is on-line at setpoint + 50 psig. So prior to the reactor trip, the TBVs would throttle open at 935 psig. Trip confirm inputs a + 125 psig bias. 885 + 125 = 1010 psig setpoint for TBVs following the reactor trip, so at 985 psig, the TBVs would be closed.

Answer C Discussion

Incorrect. First part is incorrect. Plausible since RCS pressure will decrease during this event to the point that the reactor will trip. Second part is correct.

Answer D Discussion

Correct. First part: 1RC-66 open at 100% is essentially a SBLOCA and will cause RCS pressure to decrease with little/no temperature decrease. This will cause a trip on Variable Low Pressure. Second part: Normal Turbine Header Pressure setpoint = 885 psig. Trip confirm inputs a + 125 psig bias. 885 + 125 = 1010 psig setpoint for TBVs following the reactor trip, so at 985 psig, the TBVs would be closed.

Basis for meeting the KA

Requires knowledge of value at which turbine bypass valves maintain header pressure following a reactor trip due to a PZR vapor space accident.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	ILT45 Q1

Development References
ILT45 Q1 ICS-02 Obj. 04

Student References Provided

APE008 AA2.24 - Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)

Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: (CFR: 43.5 / 45.13)

Value at which turbine bypass valve maintains header pressure after a reactor trip ....

401-9 Comments:

Remarks/Status

EPE009 2.1.20 - Small Break LOCA
EPE009 GENERIC
Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)

Given the following Unit 1 conditions:
Time = 1700

- Reactor trip from 100%

Time = 1705

- All SCMs = 0^{o}F stable
- HPI header A flow = 578 gpm stable
- 1A and 1B HPI pumps operating
- 1C HPI pump breaker failed open

1) The valve that must be opened in accordance with Rule 2 (Loss of SCM) is __(1)__.

2) In accordance with Rule 6 (HPI), once the above valve is opened the MAXIMUM <u>total</u> HPI flow is __(2)__ gpm.

Which ONE of the following completes the statements above?

A. 1. 1HP-409
2. 950

B. 1. 1HP-409
2. 750

C. 1. 1HP-410
2. 950

D. 1. 1HP-410
2. 750

General Discussion

Answer A Discussion

Correct, Rule 2 provides guidance if no flow in the "B" HPI header to open 1HP-409. In addition, if 1A & 1B HPI pumps operating with 1HP-409 open, do not exceed 950 gpm total.

Answer B Discussion

Incorrect, first part is correct. Second part is incorrect. Plausible because this is the limit on HPI flow if ONLY one LPI to HPI flow path exists while in piggy back operation.

Answer C Discussion

Incorrect, first part is incorrect. Plausible because the higher number valve would normally go to the "B" header. Second part is correct.

Answer D Discussion

Incorrect, first part is incorrect. Plausible because the higher number valve would normally go to the "B" header. Second part is incorrect. Plausible because this is the limit on HPI flow if ONLY one LPI to HPI flow path exists while in piggy back operation.

Basis for meeting the KA

Question requires evaluating the plant following a SBLOCA and determining how to adjust HPI flow (Rules 2 & 6).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT45 Q2

Development References

ILT 45 Q2
EAP-LOSCM Obj. 24
Rule 2
Rule 6

Student References Provided

EPE009 2.1.20 - Small Break LOCA

EPE009 GENERIC

Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)

401-9 Comments:

Remarks/Status

APE056 AK1.03 - Loss of Offsite Power

Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: CFR 41.8 / 41.10 / 45.3)

Definition of subcooling: use of steam tables to determine it ....

Given the following Unit 1 conditions:

Time = 1200

- Power escalation in progress
- Reactor power = 38% slowly increasing
- A Switchyard Isolation occurs

Time = 1215

- RCS pressure = 2125 psig
- Pressurizer temperature = 640°F
- Pressurizer level is stable
- All Pressurizer heaters are energized

1) At Time = 1200, an AUTOMATIC Reactor trip __(1)__ occur.

2) At Time = 1215, the pressurizer is __(2)__ .

Which ONE of the following completes the statements above?

A. 1. will
2. saturated

B. 1. will
2. subcooled

C. 1. will NOT
2. saturated

D. 1. will NOT
2. subcooled

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D

General Discussion

Answer A Discussion

Incorrect. First part is plausible since it could be true if Reactor power were > 40%. Second part is plausible since RCS presssure is below the normal pressure of 2155 psig and Pzr temperature is below normal saturated temp of 648 degrees..

Answer B Discussion

Incorrect. First part is plausible since it could be true if Reactor power were > 40%. Second part is correct

Answer C Discussion

Incorrect. First part is correct. Second part is plausible since RCS presssure is below the normal pressure of 2155 psig and Pzr temperature is below normal saturated temp of 648 degrees..

Answer D Discussion

Correct. For load rejections <40% power the reactor will not automatically trip. Saturation temperature for 2125 psig is 645.5 degrees therefore the Pzr is subcooled.

Basis for meeting the KA

KA is met since a Loss of offsite power resulted in a plant runback which has resulted in a subcooled Pzr and the candidate has to be able to determine from the steam tables if the Pzr is subcooled as well as have a grasp on the thermodynamic properties of the Pzr and RCS pressure as a result of the Pzr being subcooled.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References	Student References Provided
SAE-L020 Obj R2,3,4	

APE056 AK1.03 - Loss of Offsite Power

Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: CFR 41.8 / 41.10 / 45.3)

Definition of subcooling: use of steam tables to determine it ....

401-9 Comments:	Remarks/Status

APE015/017 AK3.05 - Reactor Coolant Pump (RCP) Malfunctions

Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow) : (CFR 41.5,41.10 / 45.6 / 45.13)

Shift of T-ave. sensors to the loop with the highest flow ....

Given the following Unit 2 conditions:
Initial conditions:

- Reactor power = 78% stable

Current conditions:

- 2A2 Reactor Coolant Pump trips

1) Controlling Tave signal will AUTOMATICALLY switch to Loop __(1)__ Tave.

2) The reason that the Controlling Tave signal is switched is due to __(2)__.

Which ONE of the following completes the statements above?

A. 1. A
2. RCP 2A2 breaker position

B. 1. B
2. RCP 2A2 breaker position

C. 1. A
2. Loop RCS flow indication

D. 1. B
2. Loop RCS flow indication

General Discussion

The stated increase in the idle loop Tc was verified on the simulator.

Answer A Discussion

1st part is is incorrect but plausible since it is one of the A loop RCP's that trip therefore it would be plausible to believe that the temperature circuit swaps to look at the affected loop.

2nd part is plausible because the input swap to Controlling Tave is based on RCS flow in the A Loop decreasing below 6.2 E6 lbm/hr. It is plausible because some RPS inputs for RC flow is based on RCP breaker position.

Answer B Discussion

1st part is correct

2nd part is plausible because the input swap to Controlling Tave is based on RCS flow in the A Loop decreasing below 6.2 E6 lbm/hr. It is plausible because some RPS inputs for RC flow is based on RCP breaker position.

Answer C Discussion

1st part is is incorrect but plausible since it is one of the A loop RCP's that trip therefore it would be plausible to believe that the temperature circuit swaps to look at the affected loop.

2nd part is correct

Answer D Discussion

Correct. The Tave circuit would monitor RCS flow and when the A Loop decreasing below 6.2 E6 lbm/hr the ciruit would swap to the other loop (Loop B).

Basis for meeting the KA

Requires knowledge of the response of the ICS Tave signal and the reason the signal swaps to the unaffected loop.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References

IC-RCI Obj. 07
ICS-04 Obj. 01
ARG

Student References Provided

APE015/017 AK3.05 - Reactor Coolant Pump (RCP) Malfunctions

Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow) : (CFR 41.5,41.10 / 45.6 / 45.13)

Shift of T-ave. sensors to the loop with the highest flow ....

401-9 Comments:

Remarks/Status

APE022 AK3.06 - Loss of Reactor Coolant Makeup

Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup: (CFR 41.5, 41.10 / 45.6 / 45.13)

RCP thermal barrier cooling . . . .

Given the following Unit 1 conditions:

- Reactor power = 100%
- All HPI pumps fail
- AP/14 (Loss of Normal HPI Makeup and/or RCP Seal Injection) initiated

1) Reactor Coolant Pump thermal barrier cooling will initially be provided by __(1)__.

2) The reason thermal barrier cooling is provided is to limit __(2)__ .

Which ONE of the following completes the statements above?

A. 1. Component Cooling
   2. damage to RCP seals

B. 1. Component Cooling
   2. temperature rise in LDST

C. 1. Low Pressure Service Water
   2. damage to RCP seals

D. 1. Low Pressure Service Water
   2. temperature rise in LDST

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General Discussion

Answer A Discussion

CORRECT. Thermal barrier cooling is provided by Component Cooling on a loss of HPI to prevent seal damage from the hot RCS flowing up through the seals.

Answer B Discussion

Incorrect. First part is correct. Second part is incorrect but plausible since seal return goes to the LDST. Therefore, on a loss of seal injection, the RCS flows up through the seal packages and goes to the LDST and the LDST has a maximum temperature limit. However, this is not the reason for thermal barrier cooling.

Answer C Discussion

Incorrect. First part is incorrect but plausible since Low Pressure Service Water provides cooling to the Reactor Coolant Pump motors. Second part is correct.

Answer D Discussion

Incorrect. First part is incorrect but plausible since Low Pressure Service Water provides cooling to the Reactor Coolant Pump motors. Second part is incorrect but plausible since seal return goes to the LDST. Therefore, on a loss of seal injection, the RCS flows up through the seal packages and goes to the LDST and the LDST has a maximum temperature limit. However, this is not the reason for thermal barrier cooling.

Basis for meeting the KA

Question requires knowledge of how RCP thermal barrier cooling is provided on a loss of Reactor Coolant Makeup (all HPI Pumps).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References

PNS-CPS Obj. 3
AP/14
EAP-APG Obj. R9

Student References Provided

APE022 AK3.06 - Loss of Reactor Coolant Makeup

Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup: (CFR 41.5, 41.10 / 45.6 / 45.13)

RCP thermal barrier cooling . . . .

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 7 7 **B**

APE025 AK2.02 - Loss of Residual Heat Removal System (RHRS)
Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: (CFR 41.7 / 45.7)
LPI or Decay Heat Removal/RHR pumps ....

Given the following Unit 1 conditions:
Time = 1200

- LPI aligned in the Normal Decay Heat Removal mode
- 1A LPI pump operating

Time = 1201

- Loss of offsite power results in Switchyard Isolation

Time = 1202

- Power restored via CT-4

1) Assuming NO operator actions, at Time = 1205 __(1)__ LPI pump(s) will be operating.

2) If NEITHER the 1A NOR the 1B LPI pumps were available at Time = 1202, manual reset of Load Shed is __(2)__ prior to starting the 1C LPI pump.

Which ONE of the following completes the statements above?

A. 1. the 1A
2. required

B. 1. the 1A
2. NOT required

C. 1. NO
2. required

D. 1. NO
2. NOT required

General Discussion

Answer A Discussion

First part is correct.

Second part is plausbile since the 1C LPI pump is not an ES component and therefore it is plausible it would respond as most other non-ES componnents and not be available until after the load shed signal had been reset.

Answer B Discussion

Correct

Since the 1A LPI pump was operating its breaker will remain closed and therefore the pump will restart once power is restored.

Pushing the Control Room MFB monitor RESET pushbuttons is not required because the signal for the 1C LPI Pump is removed 5 seconds after the Load Shed actuated if either the 1A or 1B LPI Pump is off.

Answer C Discussion

Incorrect: First part is incorrect but plausible since it would be correct if the 1C pump were operating. Additionally plausible since this is below any ECCS requirments and there are no auto start requirments for LPI pumps as it relates for DHR.
Second part is plausbile since the 1C LPI pump is not an ES component and therefore it is plausible it would respond as most other non-ES componnents and not be available until after the load shed signal had been reset.

Answer D Discussion

Incorrect: First part is incorrect but plausible since it would be correct if the 1C pump were operating. Additionally plausible since this is below any ECCS requirments and there are no auto start requirments for LPI pumps as it relates for DHR.

Second part is plausbile since the 1C LPI pump is not an ES component and therefore it is plausible it would respond as most other non-ES componnents and not be available until after the load shed signal had been reset.

Basis for meeting the KA

Requires knowledge of the relationship between a loss of DHR due to loss of power and actions or inactions required in order to restore DHR cooling following restoration of power.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	ILT44 Q6

Development References

ILT44 Q6
EL-PSL Obj 03

Student References Provided

APE025 AK2.02 - Loss of Residual Heat Removal System (RHRS)

Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: (CFR 41.7 / 45.7)

LPI or Decay Heat Removal/RHR pumps ....

401-9 Comments:

Remarks/Status

APE027 2.2.42 - Pressurizer Pressure Control System (PZR PCS) Malfunction
APE027 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 Unit 1 conditions:
Time = 1200

- Reactor in MODE 3
- RCS temperature = 450°F
- Various Breakers supplying Pzr heaters trip OPEN
- FIN24 reports 420 KW of Pressurizer heater capacity remain available

Time = 1400

- RCS temperature = 335°F stable

1) At Time = 1200, LCO 3.4.9 (Pressurizer) requirements __(1)__ met.

2) At Time = 1400, the requirements of Tech Spec 3.4.9 __(2)__ apply.

Which ONE of the following completes the statements above?

A. 1. are
2. do NOT

B. 1. are
2. do

C. 1. are NOT
2. do NOT

D. 1. are NOT
2. do

General Discussion

Answer A Discussion

1st part is correct

2nd part is incorrect but plausible since the HPI spec applicability threshold is 350 degrees..

Answer B Discussion

Correct. The LCO of TS 3.4.9 requires 400 KW of heater capacity and the spec becomes applicable at 325 degrees

Answer C Discussion

Incoxrrect. First part is plausible since 400 KW is the required number and 420 is realtively close to that number. Also, total heater capacity is over 1600 kw therefore only having 420 kw represents a significant loss in capability.

2nd part is incorrect but plausible since the HPI spec applicability threshold is 350 degrees.

Answer D Discussion

Incoxrrect. First part is plausible since 400 KW is the required number and 420 is realtively close to that number. Also, total heater capacity is over 1600 kw therefore only having 420 kw represents a significant loss in capability.

2nd part is correct.

Basis for meeting the KA

Question requires knowledge of TS entry requirements due to Pzr Heater malfunction.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References

ADM-ITS Obj. 7
TS 3.4.9
COLR
PNS-PZR Obj: 12

Student References Provided

APE027 2.2.42 - Pressurizer Pressure Control System (PZR PCS) Malfunction
APE027 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

EPE029 EK3.02 - Anticipated Transient Without Scram (ATWS)
Knowledge of the reasons for the following responses as the apply to the ATWS: (CFR 41.5 / 41.10 / 45.6 / 45.13)
Starting a specific charging pump ....

Given the following Unit 1 conditions:
Initial conditions:
- Reactor power = 100%
- 1A HPI pump operating
- 1HP-27 closed

Current conditions:
- Both Main FDW pumps trip
- Reactor power 52% decreasing
- Rule 1 (ATWS/Unanticipated Nuclear Power Production) in progress

1) In accordance with Rule 1, 1B HPI pump will be started if 1HP-27 will NOT open in order to __(1)__.

2) In accordance with Rule 6 (HPI), CRS concurrence __(2)__ be required to throttle HPI.

Which ONE of the following completes the statements above?

A. 1. maximize flow in 1A HPI header
2. will

B. 1. utilize the HPI Cross-over header
2. will

C. 1. maximize flow in 1A HPI header
2. will NOT

D. 1. utilize the HPI Cross-over header
2. will NOT

ILT48 ONS SRO NRC Examination QUESTION 9

9

B

General Discussion

Answer A Discussion

Incorrect. First part is incorrect. Plausible since HPI flow is inadequate with 1HP-27 closed and maximizing flow in 1A HPI header would increase the boration. However, the intent of the EOP is to provide flow in both headers. Second part is correct.

Answer B Discussion

CORRECT. The Standby HPI Pump (1B in this case) will be started per Rule 1 if 1HP-27 will not open and 1HP-409 (HPI cross-over) will be opened to provide flow in both headers.. CRS concurrence is required to throttle HPI when emergency boration is in effect.

Answer C Discussion

Incorrect. First part is incorrect. Plausible since HPI flow is inadequate with 1HP-27 closed and maximizing flow in 1A HPI header would increase the boration. However, the intent of the EOP is to provide flow in both headers.. Second part is incorrect. Plausible since it would be correct if not in emergency boration.

Answer D Discussion

Incorrect. First part is correct. Second part is incorrect. Plausible since it would be correct if not in emergency boration.

Basis for meeting the KA

The question requires knowledge of a reason the Standby HPI Pump is started during an ATWS.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

EAP-UNPP Obj. R12
EAP-UNPP (Rule 1)
Rule 1

Student References Provided

EPE029 EK3.02 - Anticipated Transient Without Scram (ATWS)

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

Starting a specific charging pump ....

401-9 Comments:

Remarks/Status

EPE038 EA2.16 - Steam Generator Tube Rupture (SGTR)
Ability to determine or interpret the following as they apply to a SGTR : (CFR 43.5 / 45.13)
Actions to be taken if S/G goes solid and water enters steam line ....

Given the following Unit 1 conditions:

- Steam Generator Tube Rupture in 1B SG
- Tcold = 490°F slowly decreasing
- RB pressure = 3.1 psig slowly decreasing
- RB temperature = 203°F slowly decreasing
- 1B SG pressure = 606 psig slowly decreasing
- 1B SG Full Range level = 54% slowly increasing

1) 1B SG level __(1)__ reached the level at which water can enter the Main Steam lines.

2) Assuming 1B SG has reached the level of water in the Main Steam line, __(2)__.

Which ONE of the following completes the statements above?

REFERENCE PROVIDED

A. 1. has
2. discontinue steaming the affected SG

B. 1. has NOT
2. discontinue steaming the affected SG

C. 1. has
2. maximize steaming of the unaffected SG even if TS cooldown rates are exceeded

D. 1. has NOT
2. maximize steaming of the unaffected SG even if TS cooldown rates are exceeded

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General Discussion

Answer A Discussion

CORRECT. First part is correct. Per EOP Encl. 5.21 (reference provided), the level at which water can enter the steam line is 52.7% Full Range. 1B SG level = 54% Full Range, therefore water has reached the level of water in the steam line. Second part is correct. Once the water in MS line level is reached, steaming is discontinued per the SGTR tab.

Answer B Discussion

Incorrect. First part is incorrect. Plausible since the candidate has to determine level from Encl. 5.21 using RB temperature, pressure, and SG pressure. An error on any of the three parameters can cause the candidate to determine an incorrect level, which could indicate water has reached the level to enter the main steam lines. Second part is correct.

Answer C Discussion

Incorrect. First part is correct. Second part is incorrect. Plausible since steaming the SG even if cooldown rates are exceeded is guidance provided in the SGTR Tab if the SG is approaching overfill conditions and maximizing steaming of the unaffected SG would decrease RCS temp and pressure, which would decrease the leak size in the 1B SG.

Answer D Discussion

Incorrect. First part is incorrect. Plausible since the candidate has the determine level from Encl. 5.21 using RB temperature, pressure, and SG pressure. An error on any of the three parameters can cause the candidate to determine an incorrect level, which could indicate water has reached the level to enter the main steam lines. Second part is incorrect. Plausible since steaming the SG even if cooldown rates are exceeded is guidance provided in the SGTR Tab if the SG is approaching overfill conditions and maximizing steaming of the unaffected SG would decrease RCS temp and pressure, which would decrease the leak size in the 1B SG.

Basis for meeting the KA

The question requires knowledge of actions to be taken in accordance with the SGTR Tab when SG level reaches the point at which water can enter the MS line.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References
EAP-SGTR Obj. R 15 & 17 SGTR EOP Encl. 5.21

Student References Provided
EOP Encl. 5.21

EPE038 EA2.16 - Steam Generator Tube Rupture (SGTR)

Ability to determine or interpret the following as they apply to a SGTR : (CFR 43.5 / 45.13)

Actions to be taken if S/G goes solid and water enters steam line ....

401-9 Comments:	Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 11 11 **A**

APE054 AK1.02 - Loss of Main Feedwater (MFW)
Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW): (CFR 41.8 / 41.10 / 45.3)
Effects of feedwater introduction on dry S/G ....

Given the following Unit 2 conditions:

- Loss of Heat Transfer has occurred
- Unit 2 TDEFWP is now available to feed the Steam Generators
- 2A SG level = 8" XSUR slowly decreasing
- 2A SG pressure = 412 psig slowly decreasing
- 2B SG level = 5" XSUR slowly decreasing
- 2B SG pressure = 385 psig slowly decreasing

In accordance with Rule 7 (Steam Generator Feed Control), the MAXIMUM initial feed rate allowed to EACH Steam Generator is limited to __(1)__ gpm in order to prevent excessive __(2)__ to the Steam Generator tubes.

Which ONE of the following completes the statement above?

A. 1. 100
2. excessive stresses

B. 1. 100
2. flow induced vibration damage

C. 1. 50
2. excessive stresses

D. 1. 50
2. flow induced vibration damage

General Discussion

Answer A Discussion

Correct. Rule 7 limits flow to each affected Steam Generator to 100 gpm if the SG is dry. Level below 12" along with low and decreasing SG pressure indicates that the SG is dry, The EOP TBD explains that the 100 gpm flow limit is based on protecting the SG tubes from excessive stresses.

Answer B Discussion

Incorrect. First part is correct.

Second part is plausible since it would be correct if the SG already had Heat Trasfer established since that is the basis for flow limits for a SG 'with heat transfer.

Answer C Discussion

Incorrect. First part is plausible under the misconception that the 100 gpm is a total flow limit instead of a limit to each SG. 100 gpm total flow is correct if feeding with SSF ASWP.
Second part is correct.

Answer D Discussion

Incorrect. First part is plausible under the misconception that the 100 gpm is a total flow limit instead of a limit to each SG. 100 gpm total flow is correct if feeding with SSF ASWP.
Second part is plausible since it would be correct if the SG already had Heat Trasfer established since that is the basis for flow limits for a SG 'with heat transfer.

Basis for meeting the KA

Requires knowledge of the operational implications of not adhering to the flow limits established when introducing feed to a dry SG with no heat transfer.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT46 Q10

Development References

ILT46 Q10
EAP-LOHT Obj R27
Rule 7
Rule 7 lesson plan

Student References Provided

APE054 AK1.02 - Loss of Main Feedwater (MFW)

Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW): (CFR 41.8 / 41.10 / 45.3)

Effects of feedwater introduction on dry S/G ....

401-9 Comments:

Remarks/Status

EPE055 EA1.07 - Loss of Offsite and Onsite Power (Station Blackout)
Ability to operate and monitor the following as they apply to a Station Blackout: (CFR 41.7 / 45.5 / 45.6)
Restoration of power from offsite ....

Given the following Unit 1 conditions:
Initial conditions:

- Reactor power = 100%
- ACB-4 closed
- A Switchyard Isolation occurs

Current conditions:

- Keowee Unit 2 Emergency lockout
- 230 KV Yellow Bus Differential lockout

Main Feeder buses will be energized __(1)__ from __(2)__.

Which ONE of the following completes the statement above?

A. 1. manually
2. CT-4

B. 1. automatically
2. CT-4

C. 1. manually
2. CT-5

D. 1. automatically
2. CT-5

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General Discussion

Answer A Discussion

Correct. Since ACB-4 is closed, KHU-2 is tied to the underground (CT-4) and KHU-1 is tied to the Overhead. When the event occurs, KHU-1 is not capable of going through the overhead due to the Yellow Bus Differential lockout. KHU-2 is not available due to the emergency lockout. Encl 5.38 will direct lining up the available KHU unit (KHU-1) to the MFBs through CT-4.

Answer B Discussion

Incorrect because there are no automatic means to line up the KHU aligned to the Overhead path (KHU-1) to power the Main Feeder Busses via CT-4 under the given conditions. It is plausible because if ACB 3 were closed instead of ACB-4, it would be correct.

Answer C Discussion

Incorrect because Encl 5.38 will direct powering the Main Feeder Busses from KHU-1 through CT-4. It is plausible because if neither KHU were available, it would be correct.

Answer D Discussion

Incorrect because Encl 5.38 will direct powering the Main Feeder Busses from KHU-1 through CT-4. It is plausible because if the Standby Busses were energized from CT-5 at the time of the event, it would be correct.

Basis for meeting the KA

The applicant must be able to operate and monitor equipment associated with the restoration of power from off-site following a blackout.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	ILT39 Q11

Development References

ILT39 Q11
EL-PSL Obj. 07
ARG SA-3/C3

Student References Provided

EPE055 EA1.07 - Loss of Offsite and Onsite Power (Station Blackout)

Ability to operate and monitor the following as they apply to a Station Blackout: (CFR 41.7 / 45.5 / 45.6)

Restoration of power from offsite ....

401-9 Comments:

Remarks/Status

APE056 2.4.20 - Loss of Offsite Power

APE056 GENERIC

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

Given the following Unit 1 conditions:

- Station Blackout has occurred

Concerning the TDEFWP…

1) Oil cooling water will be supplied by __(1)__.

2) the MINIMUM steam pressure that will ensure proper operation of the pump is __(2)__ psig.

Which ONE of the following completes the statements above?

A. 1. HPSW
   2. 250

B. 1. HPSW
   2. 300

C. 1. LPSW
   2. 250

D. 1. LPSW
   2. 300

General Discussion

Answer A Discussion

Correct. The TDEFDWP Oil Cooler is normally supplied by CCW by an AC powered oil cooling water pump that starts when MS-93 (Steam Admission Valve) opens. HPSW provides backup cooling. Since the unit has experienced a blackout, the CCW supply will not be available and HPSW will supply the oil cooler.
The minimum steam pressure that will ensure proper operation of the TDEFDW Pump is 250 psig.

Answer B Discussion

Incorrect. First part is correct. Second part is incorrect. Plausible since this is the normal pressure for the Auxiliary Steam Header which is one of the steam supplies for the TDEFDW Pump. The other is Main Steam.

Answer C Discussion

Incorrect. First part is incorrect. Plausible since LPSW supplies cooling water to the MDEFDW Pumps. Second part is correct.

Answer D Discussion

Incorrect. First part is incorrect. Plausible since LPSW supplies cooling water to the MDEFDW Pumps. Second part is incorrect. Plausible since this is the normal pressure for the Auxiliary Steam Header which is one of the steam supplies for the TDEFDW Pump. The other is Main Steam.

Basis for meeting the KA

The question requires knowledge of the EOP Caution related to minimum steam pressure required for proper TDEFDW Pump operation.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

CF-EF Obj. 22 & 55
EOP BO Tab

Student References Provided

APE056 2.4.20 - Loss of Offsite Power
APE056 GENERIC
Knowledge of the operational implications of EOP warnings, cautions, and notes. (CFR: 41.10 / 43.5 / 45.13)

401-9 Comments:

Remarks/Status

APE057 AA1.06 - Loss of Vital AC Electrical Instrument Bus

Ability to operate and / or monitor the following as they apply to the Loss of Vital AC Instrument Bus: (CFR 41.7 / 45.5 / 45.6)

Manual control of components for which automatic control is lost ....

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%
- SASS in Manual while SPOC repairs Pressurizer Level 3 transmitter
- 1HP-120 in AUTO selected to Pressurizer Level 1

Current conditions:

- Vital power to ICCM Train A fails

Which ONE of the following describes Pressurizer level control with 1HP-120?

A. Selecting Pressurizer Level 2 and depressing the AUTO pushbutton on 1HP-120 are required to restore automatic control at setpoint

B. Selecting Pressurizer Level 2 ONLY will restore automatic control at setpoint

C. Manual control using 1HP-120 Bailey controller is all that is available

D. Additional actions are NOT required since Automatic control at setpoint is retained

---

General Discussion

Answer A Discussion

Incorrect. ICCM Train A feeds both Pzr level 1 & 2. ICCM Train B feeds Pzr level 3. It is plausible to believe that since ICCM Train A feeds Pzr level 1 then ICCM Train B feeds Pzr level 2. Under this misconception it is plausible to believe that 1HP-120 would trip to Hand when power is lost to Pzr level 1 since there are multiple bailey control stations that trip to hand under various conditions.

Answer B Discussion

Incorrect. ICCM Train A feeds both Pzr level 1 & 2. ICCM Train B feeds Pzr level 3. It is plausible to believe that since ICCM Train A feeds Pzr level 1 then ICCM Train B feeds Pzr level 2 which would lead choosing this as the correct answer

Answer C Discussion

Correct. ICCM Train A feeds both Pzr level 1 & 2. ICCM Train B feeds Pzr level 3. With Pzr level 3 unavailable, if ICCM Train A fails, all auto control is lost, therefore only using 1HP-120 in hand would be correct.

Answer D Discussion

Incorrect. Plausible since this would be correct if SASS were in Auto.

Basis for meeting the KA

Requires the ability to both manually operate and monitor manual control of 1fHP-120 following a loss of Vital Power to ICCM Train A.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT44 Q14

Development References

ILT44 Q14
PNS-PZR Obj. 20 & 21
OP/1/A/1105/012

Student References Provided

APE057 AA1.06 - Loss of Vital AC Electrical Instrument Bus

Ability to operate and / or monitor the following as they apply to the Loss of Vital AC Instrument Bus: (CFR 41.7 / 45.5 / 45.6)

Manual control of components for which automatic control is lost ....

401-9 Comments:

Remarks/Status

APE065 AA1.01 - Loss of Instrument Air
Ability to operate and / or monitor the following as they apply to the Loss of Instrument Air: (CFR 41.7 / 45.5 / 45.6)
Remote manual loaders ....

Given the following Unit 1 conditions:
Time = 0800:
- Reactor power = 100%
- Instrument Air pressure decreasing
- AP/22 (Loss of Instrument Air) initiated

Time = 0810:
- 1B Main Steam Line Break in containment
- Reactor Building pressure 3.2 psig slowly increasing
- Instrument Air and Auxiliary Instrument Air pressure lost

At Time = 0810…

1) In accordance with Rule 7 (SG Feed Control), 1A SG will be controlled at __(1)__ inches XSUR level.

2) 1FDW-315 __(2)__ be operated from the Control Room.

Which ONE of the following completes the statements above?

A. 1. 30
2. can

B. 1. 30
2. can NOT

C. 1. 60
2. can

D. 1. 60
2. can NOT

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General Discussion

Answer A Discussion

Incorrect. First part is incorrect. Plausible since it would be correct if RB pressure were < 3 psig. Second part is correct. 1FDW-315/316 have a nitrogen backup that will allow normal operation from the controllers in the Control Room for approximately 2 hours following a loss of IA and AIA.

Answer B Discussion

Incorrect. First part is incorrect. Plausible since it would be correct if RB pressure were < 3 psig. Second part is incorrect. Plausible since it would be correct without the nitrogen backup, or once the nitrogen supply is depleted.

Answer C Discussion

CORRECT. With degraded containment (RB pressure >/= 3 psig), abnormal containment (acc) levels apply and 30 inches is added to the SG level setpoints when on EFDW. Therefore, the SG acc level setpoint when on EFDW with RCPs operating is 60" XSUR. The operator must maintain the acc level with 1FDW-315/316 in manual since no auto control is available. Second part is correct.

Answer D Discussion

Incorrect. First part is correct. Second part is incorrect. Plausible since it would be correct without the nitrogen backup, or once the nitrogen supply is depleted.

Basis for meeting the KA

The question requires knowledge of the ability to manually operate 1FDW-315/316 controllers during a loss of Instrument Air.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

CF-EF Obj. 41
EAP-LOHT Rule 7 Obj. R27
Rule 7

Student References Provided

APE065 AA1.01 - Loss of Instrument Air

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

Remote manual loaders ....

401-9 Comments:

Remarks/Status

APE077 AA2.06 - Generator Voltage and Electric Grid Disturbances

Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: (CFR: 41.5 and 43.5 / 45.5, 45.7, and 45.8)

Generator frequency limitations..

Given the following Unit 1 conditions:

- Reactor power = 55% stable
- AP/34 (Generator Grid Disturbance) in progress
- Generator Frequency = 57 Hz
- Generator Output = 850 MWe and (+) 450 MVARs
- Generator Hydrogen Pressure = 60 psig
- Generator Output Voltage = 18.2 KV

1) In accordance with AP/34 the Generator output __(1)__ within the limits of the Generator Capability Curve.

2) Manually tripping the Main Turbine__(2)__ required.

Which ONE of the following completes the statements above?

REFERENCE PROVIDED

A. 1. is
2. is

B. 1. is NOT
2. is

C. 1. is
2. is NOT

D. 1. is NOT
2. is NOT

General Discussion

Answer A Discussion

1st part is correct. The generator output is within the limits of the Generator Capability Curve (AP/34 Encl. 5.1).

2nd part is correct. An automatic main turbine trip on low frequence should have occurred at 57.6 hz and since it did not occur (based on stem conditions) it should be manually tripped.

Answer B Discussion

1st part is incorrect. It is plausible since it would be correct if power factor were leading or if Gen H2 pressure were lower.

2nd part is correct. A Reactor trip is required by AP/34 due to frequency being < 57.6 HZ and reactor power > 50%.

Answer C Discussion

1st part is correct. The generator output is within the limits of the Generator Capability Curve (AP/34 Encl. 5.1).

2nd part is incorrect because a reactor trip is required due to frequency being low. It is plausible since it would be correct if reactor power were </= 50%.

Answer D Discussion

1st part is incorrect. It is plausible since it would be correct if power factor were leading or if Gen H2 pressure were lower.

2nd part is incorrect because a reactor trip is required due to frequency being low. It is plausible since it would be correct if reactor power were </= 50%.

It is plausible to not be within the generator capability curve and not require a reactor trip because if parameters were such that you were not within the capability curve but were < 50% power, it would be correct.

Basis for meeting the KA

The question requires the ability to determine if a Generator frequency limit has been exceeded during an electric grid disturbance and whehter a reactor trip is warranted.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	ILT44 Q16

Development References

ILT44 Q16
EAP-APG (AP/34) Obj. R4 and R7
AP/1/A/1700/034

Student References Provided

AP/34 Encl. 5.1

APE077 AA2.06 - Generator Voltage and Electric Grid Disturbances

Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: (CFR: 41.5 and 43.5 / 45.5, 45.7, and 45.8)

Generator frequency limitations...

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 17 | 17 | D

BWE04 EK2.2 - Inadequate Heat Transfer

Knowledge of the interrelations between the (Inadequate Heat Transfer) 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:

- Loss of Heat Transfer exists due to the loss of ALL FDW sources
- HPI Forced Cooling in progress
- RCS pressure = 2210 psig slowly decreasing
- Pzr Level = 380 inches increasing
- Core SCM = 56°F increasing

In accordance with Rule 6 (HPI), HPI flow __(1)__ be throttled because __(2)__.

Which ONE of the following completes the statement above?

A. 1. may NOT
2. RCS pressure is decreasing

B. 1. may NOT
2. CETCs are increasing

C. 1. may
2. Pzr Level is increasing

D. 1. may
2. CETCs are decreasing

General Discussion

Answer A Discussion

Incorrect: First part is incorrect. Criteria for throttling HPI during HPI cooling is based on Core SCM >0 and CETC decreasing. Plausible since decreasing RCS pressure can be an indication of overcooling.

Answer B Discussion

Incorrect: First part is incorrect. Criteria for throttling HPI during HPI cooling is based on Core SCM >0 and CETC decreasing. Plausible if correlation between increasing Core SCM and slowly decreasing pressure is not recognized as indication that CETC temperatures are decreasing.

Answer C Discussion

Incorrect: First part is correct. Second part is incorrect but plausible in that Pzr level increasing is part of the HPI throttling criteria if NOT in HPI F/C.

Answer D Discussion

Correct: Criteria for throttling HPI during HPI cooling is based on Core SCM >0 and CETC decreasing. Core SCM increasing with RCS pressure slowly decreasing indicates that CETC temperatures are decreasing.

Basis for meeting the KA

Requires knowledge of heat removal systems (HPI FC) and their operation during a loss of heat transfer event.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	2009 Q18

Development References

2009 Q18
EAP-HPICD Obj. R3
Rule 6

Student References Provided

BWE04 EK2.2 - Inadequate Heat Transfer

Knowledge of the interrelations between the (Inadequate Heat Transfer) 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

BWE05 EK1.1 - Excessive Heat Transfer
Knowledge of the operational implications of the following concepts as they apply to the (Excessive Heat Transfer)
(CFR: 41.8 / 41.10 / 45.3)
Components, capacity, and function of emergency systems.

Given the following Unit 1 conditions:
Initial conditions:

- Reactor power = 100%
- 1A Main Steam Line Break occurs

Current conditions:

- Reactor has tripped
- RCS Tave = 544°F slowly increasing
- 1A SG pressure = 0 psig
- 1B SG pressure = 990 psig slowly increasing

1) The Turbine Driven Emergency Feedwater Pump (TDEFDWP) is __(1)__.

2) The TDEFDWP can be __(2)__ AFIS is reset.

Which ONE of the following completes the statements above?

A. 1. operating
2. secured before

B. 1. operating
2. secured ONLY after

C. 1. NOT operating
2. started before

D. 1. NOT operating
2. started ONLY after

ILT48 ONS SRO NRC Examination QUESTION 18

18

C

General Discussion

Answer A Discussion

Incorrect: Plausible since 1FDW-315 is closed in first step of Rule 5. This makes it plausible that AFIS would not secure the TDEFDWP so that it would be available to feed the B SG if needed.

Answer B Discussion

Incorrect: Plausible since 1FDW-315 is closed in first step of Rule 5. This makes it plausible that AFIS would not secure the TDEFDWP so that it would be available to feed the B SG if needed. Second part is plausible since many components require manual action other than just turning switch to re-position following a safety system actuation (ex: ES components).

Answer C Discussion

CORRECT: The TDEFDWP control switch will override the AFIS interlock to close TO-145. TO-145 blocks the hydraulic oil supply to MS-95 therefore stopping steam supply to the TDEFDWP. The TDEFDWP switch overrides the AFIS signal and allows the operator to restart the TDEFDWP as necessary to feed Steam Generators without resetting the AFIS signal.

Answer D Discussion

Incorrect: TDEFDWP would be off. Second part is plausible since many components require manual action other than just turning switch to re-position following a safety system actuation (ex: ES components).

Basis for meeting the KA

The question requires knowledge of the operational implications of an AFIS actuation (emergency system) on the operation of the TDEFDWP (component) during an Excessive Heat Transfer event.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	2010A Q18

Development References

2010A Q18
CF-EF Obj. 30

Student References Provided

BWE05 EK1.1 - Excessive Heat Transfer

Knowledge of the operational implications of the following concepts as they apply to the (Excessive Heat Transfer)
(CFR: 41.8 / 41.10 / 45.3)

Components, capacity, and function of emergency systems.

401-9 Comments:

Remarks/Status

APE001 AK3.01 - Continuous Rod Withdrawal

Knowledge of the reasons for the following responses as they apply to the Continuous Rod Withdrawal : (CFR 41.5,41.10 / 45.6 / 45.13)
Manually driving rods into position that existed before start of casualty ...

Given the following Unit 1 conditions:

- Reactor power = 75% stable
- Instrument failure results in rod withdrawal
- ICS is taken to hand during Plant Transient Response

In accordance with OMP 1-18 (Implementation Standard During Abnormal and Emergency Events)…

1) In order to declare the plant stable, __(1)__ power must be less than or equal to the pre-transient level.

2) When inserting control rods during PTR, the criteria to stop the initial control rod insertion is when RCS pressure and Tave __(2)__.

Which ONE of the following completes the statements above?

A. 1. NI
2. stop increasing

B. 1. NI
2. return to the pre-transient values

C. 1. Core Thermal
2. stop increasing

D. 1. Core Thermal
2. return to the pre-transient values

General Discussion

Answer A Discussion

Incorrect. First part is plausible since NI's are what provide all reactor protection. Second part is correct.

Answer B Discussion

Incorrect. First part is plausible since NI's are what provide all reactor protection. Second part is plausible since Core thermal power is reduced to less than or equal to the pre-transient value during PTR.

Answer C Discussion

Correct. During Plant Transient Response (PTR), OMP 1-18 direction is to reduce Core Thermal Power to a value less than or equal to the pre-transient value and insert control rods when RCS pressure and Tave are increasing due to the heat balance mismatch.

Answer D Discussion

Incorrect. First part is correct. Second part is plausible since Core thermal power is reduced to less than or equal to the pre-transient value during PTR.

Basis for meeting the KA

This question requires knowledge of the reason control rods would be inserted during the casualty. The reason rods would be inserted at this power level would be to gain control of increasing RCS pressure (generally associated with increasing temperature).. At ONS, the focus is not returning rods to the pre-transient position but it is placing control rods in a position that match FDW flow and that is determined by RCS pressure/temperature response.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

ADM-OMP Obj R10
OMP 1-18 Att J

Student References Provided

APE001 AK3.01 - Continuous Rod Withdrawal

Knowledge of the reasons for the following responses as they apply to the Continuous Rod Withdrawal : (CFR 41.5,41.10 / 45.6 / 45.13)

Manually driving rods into position that existed before start of casualty ...

401-9 Comments:

Remarks/Status

Preview

New KA

ILT48 ONS SRO NRC Examination QUESTION 20 20 **A**

APE003 AA1.05 - Dropped Control Rod
Ability to operate and / or monitor the following as they apply to the Dropped Control Rod: (CFR 41.7 / 45.5 / 45.6)
Reactor power - turbine power ....

Given the following Unit 1 conditions:
Initial conditions:

- Reactor power = 68% increasing

Current conditions:

- Control Rod group 6 Rod 6 = 0% withdrawn

1) An ICS Asymmetric Rod Runback will reduce power to __(1)__ .

2) If occurring, depressing the HOLD pushbutton on the LCP __(2)__ stop an ICS Asymmetric Rod Runback.

Which ONE of the following completes the statements above?

A. 1. 55%
2. will

B. 1. 55%
2. will NOT

C. 1. 60%
2. will

D. 1. 60%
2. will NOT

General Discussion

Answer A Discussion

Correct:
1st part is correct. ICS is set to reduce reactor power to 55% upon receiving an Asymmetric Rod Runback.

2nd part is correct. Depressing the HOLD button will stop an Asymmetric Rod Runback.

Answer B Discussion

1st part is correct. ICS is set to reduce reactor power to 55% upon receiving an Asymmetric Rod Runback.

2nd part is incorrect because depressing HOLD will stop an Asymmetric Rod Runback. It is plausible since depressing the HOLD pushbutton will NOT stop any other runback.

Answer C Discussion

1st part is incorrect because power will be reduced to 55% upon receiving an Asymmetric Rod Runback. It is plausible because the TS setpoint for the Asymmetric Rod Runback is <=60%.

2nd part is correct. Depressing the HOLD button will stop an Asymmetric Rod Runback.

Answer D Discussion

1st part is incorrect because power will be reduced to 55% upon receiving an Asymmetric Rod Runback. It is plausible because the TS setpoint for the Asymmetric Rod Runback is <=60%.

2nd part is incorrect because depressing HOLD will stop an Asymmetric Rod Runback. It is plausible since depressing the HOLD pushbutton will NOT stop any other runback.

Basis for meeting the KA

The question requires knowledge of if/how power changes due to a dropped control rod.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT39 Q20

Development References

ILT39 Q20
IC-CRI Obj. 11

Student References Provided

APE003 AA1.05 - Dropped Control Rod
Ability to operate and / or monitor the following as they apply to the Dropped Control Rod: (CFR 41.7 / 45.5 / 45.6)
Reactor power - turbine power ....

401-9 Comments:

Remarks/Status

APE005 AK1.03 - Inoperable/Stuck Control Rod

Knowledge of the operational implications of the following concepts as they apply to Inoperable / Stuck Control Rod: (CFR 41.8 / 41.10 / 45.3)

Xenon transient ....

Given the following Unit 1 conditions:

Initial conditions:

- Reactor shutdown in progress
- Reactor power = 58% decreasing
- Control Rod 3 in Group 7 mechanically bound

Current conditions:

- 1SA-2/B-10 (CRD Asymmetric Rod Position Error) actuates
- Power decrease stopped
- Reactor power = 50% stable

1) Over the next 2 hours, Control Rod 3 in Group 7 __(1)__ become closer to the Group 7 average position.

2) If a Reactor trip occurs, Group 7 in limit __(2)__ be indicated.

Which ONE of the following completes the statements above?

ASSUME NO OPERATOR ACTIONS

A. 1. will
   2. will

B. 1. will NOT
   2. will

C. 1. will
   2. will NOT

D. 1. will NOT
   2. will NOT

ILT48 ONS SRO NRC Examination QUESTION 21

21 **A**

General Discussion

Answer A Discussion

CORRECT. With the power decrease stopped, Xenon will build in from the downpower causing Group 7 control rods to withdraw and become closer to the Rod 3 position. The Group in limit is received when any rod in that group reaches the limit switch.

Answer B Discussion

Incorrect. First part is incorrect. Plausible since it is a common error to confuse control rod movement with changes in Xenon and boron. Second part is correct. The Group in limit is received when any rod in that group reaches the limit switch.

Answer C Discussion

Incorrect. First part is correct. Second part is incorrect. Plausible to believe the Group in limit indication would require each rod in the group to be at the in limit.

Answer D Discussion

Incorrect. First part is incorrect. Plausible since it is a common error to confuse control rod movement with changes in Xenon and boron. Second part is incorrect. Plausible to believe the Group in limit indication would require each rod in the group to be at the in limit.

Basis for meeting the KA

Question requires understanding the operational implications of long term operation with a misaligned control rod.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References	Student References Provided
IC-CRI Obj. 07 RT-RBC Obj. 03	

APE005 AK1.03 - Inoperable/Stuck Control Rod

Knowledge of the operational implications of the following concepts as they apply to Inoperable / Stuck Control Rod: (CFR 41.8 / 41.10 / 45.3)

Xenon transient ....

401-9 Comments:	Remarks/Status

APE028 2.4.11 - Pressurizer (PZR) Level Control Malfunction
APE028 GENERIC
Knowledge of abnormal condition procedures. (CFR: 41.10 / 43.5 / 45.13)

Given the following Unit 1 conditions:

- Reactor power = 100%
- Pressurizer level = 210” slowly decreasing
- 1HP-120 (RC VOLUME CONTROL) failed CLOSED
- AP/14 (Loss of Normal HPI Makeup and/or RCP Seal Injection) initiated

In accordance with AP/14...

1) RCS makeup is <u>initially</u> provided by throttling __(1)__.

2) pressurizer level is maintained at a MINIMUM of __(2)__ inches.

Which ONE of the following completes the statements above?

A. 1. 1HP-26
2. 200

B. 1. 1HP-26
2. 80

C. 1. 1HP-122 (RC VOLUME CONTROL BYPASS)
2. 200

D. 1. 1HP-122 (RC VOLUME CONTROL BYPASS)
2. 80

General Discussion

Answer A Discussion

CORRECT: AP/14 directs throttling makeup through HP-26 to maintain PZR >200". If HP-26 fails, AO will locally open HP-122 (HP-120 bypass).

Answer B Discussion

Incorrect: First part is correct. Second part is plausible since 80" is the pressurizer level required to maintain pressurizer heater operability. Rule 6 allows throttling provided pressurizer level is increasing and with the 80" heater cutoff it could be a reasonable misconception that 80" is the low level limit.

Answer C Discussion

Incorrect: First part is incorrect. First part is plausible since 1HP-122 would be correct if 1HP-26 would not open. Second part is correct.

Answer D Discussion

Incorrect: Both parts are incorrect. First part is plausible since 1HP-122 would be correct if 1HP-26 would not open. Second part is plausible since 80" is the pressurizer level required to maintain pressurizer heater operability. Rule 6 allows throttling provided pressurizer level is increasing and with the 80" heater cutoff it could be a reasonable misconception that 80" is the low level limit.

Basis for meeting the KA

Question requires knowledge of Abnormal Procedure (AP/14) pertaining to a Pzr level control malfunction.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	2010A Q19

Development References

2010A Q19
EAP-APG Obj. R9
AP/14

Student References Provided

APE028 2.4.11 - Pressurizer (PZR) Level Control Malfunction
APE028 GENERIC
Knowledge of abnormal condition procedures. (CFR: 41.10 / 43.5 / 45.13)

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 23 23 **D**

APE037 AK1.02 - Steam Generator (S/G) Tube Leak

Knowledge of the operational implications of the following concepts as they apply to Steam Generator Tube Leak: CFR 41.8 / 41.10 / 45.3)

Leak rate vs. pressure drop ....

Given the following Unit 1 conditions:

- SGTR in the 1A SG
- SGTR tab in progress
- RCS temperature = 540°F decreasing
- 1TA and 1TB are de-energized

In accordance with the SGTR tab…

1) Core SCM is decreased during cooldown to __(1)__.

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

Which ONE of the following completes the statements above?

A. 1. minimize tensile stress on the 1A SG
2. Pzr spray

B. 1. minimize tensile stress on the 1A SG
2. the PORV

C. 1. reduce the primary to secondary leak rate
2. Pzr spray

D. 1. reduce the primary to secondary leak rate
2. the PORV

ILT48 ONS SRO NRC Examination QUESTION 23 23 **D**

General Discussion

Answer A Discussion

Incorrect: First part is incorrect but plausible since SCM is reduced when a SG is isolated in the EHT tab to reduce tensile stress. It is not the reason in this case. Second part is incorrect but plausible since it would be correct if the RCP that provides PZR spray was operating.

Answer B Discussion

Incorrect: First part is incorrect but plausible since SCM is reduced when a SG is isolated in the EHT tab to reduce tensile stress. It is not the reason in this case. Second part is correct.

Answer C Discussion

Incorrect. First part is correct. Second part is incorrect but plausible since it would be correct if the RCP that provides PZR spray was operating.

Answer D 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
Using the PORV is a strategy used in the SGTR tab to reduce SCM in this case because the RCPs are not operating since 1TA and 1TB are de-energized.

Basis for meeting the KA

Requires knowing that SCM is reduced when a SG has a tube rupture in order to decrease RCS pressure which in turn decreases the leak rate.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT41 Q8

Development References

Q23 ILT41 Q8
EAP-SGTR Obj. R1 and 4
EOP SGTR

Student References Provided

APE037 AK1.02 - Steam Generator (S/G) Tube Leak

Knowledge of the operational implications of the following concepts as they apply to Steam Generator Tube Leak: CFR 41.8 / 41.10 / 45.3)

Leak rate vs. pressure drop ....

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 24 24 **B**

APE059 AA1.01 - Accidental Liquid Radioactive-Waste Release
Ability to operate and / or monitor the following as they apply to the Accidental Liquid Radwaste Release: (CFR 41.7 / 45.5 / 45.6)
Radioactive-liquid monitor ....

Given the following Unit 1 conditions:

- AP/31 (Primary to Secondary Leakage) in progress

1) As a result of Primary to Secondary leakage, the Unit 1/2 Turbine Building Sump (TBS) pump breakers are opened __(1)__.

2) A sustained loss of power to 1RIA-54 will trip BOTH Turbine Building Sump Pumps __(2)__.

Which ONE of the following completes the statements above?

A. 1. ONLY when RIA-54 is in alarm or inoperable
2. after a 2 minute timer

B. 1. ONLY when RIA-54 is in alarm or inoperable
2. immediately

C. 1. ANYTIME AP/31 (Primary to Secondary Leakage) has been initiated
2. after a 2 minute timer

D. 1. ANYTIME AP/31 (Primary to Secondary Leakage) has been initiated
2. immediately

General Discussion

Answer A Discussion

1st part is correct. AP/31 directs opening the TB sump pump breakers if either 1RIA-54 is in alarm OR inoperable.

2nd part is plausible because there is a 2 minute timer associated with low sample pump flow that provides for an automatic backwash of the strainer on high strainer DP. It is plausible to believe it applies here since most SLC requirements for RIA's have a provision to allow in progress releases to continue on loss of the associated RIA's which makes a 2 minute timer to allow power to be restored additionally plausible.

Answer B Discussion

1st part is correct. AP/31 directs opening the TB sump pump breakers if either 1RIA-54 is in alarm OR inoperable.

2nd part is correct,

Answer C Discussion

1st part is incorrect because AP/31 does not direct opening TB Sump Pump breakers unless 1RIA-54 is in High Alarm or inoperable. It is plausible because the contaminated water from the secondary will eventually make it into the TB Sump so it would be conservative to open the breakers before the alarm is received.

2nd part is incorrect because the sump pumps will trip immediately. It is plausible because there is a 2 minute timer associated with low sample pump flow that provides for an automatic backwash of the strainer on high strainer DP. It is plausible to believe it applies here since most SLC requirements for RIA's have a provision to allow in progress

Answer D Discussion

1st part is incorrect because AP/31 does not direct opening TB Sump Pump breakers unless 1RIA-54 is in High Alarm or inoperable. It is plausible because the contaminated water from the secondary will eventually make it into the TB Sump so it would be conservative to open the breakers before the alarm is received.

2nd part is correct. A loss of power to RIA=54 will automatically trip both TBS pump breakers.

Basis for meeting the KA

Requires the ability to monitor proper operation of the Radioactive Liquid Monitor (RIA-54) to prevent an accidental release.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	MODIFIED	ILT44 Q50

Development References

ILT44 Q50
RAD-RIA Obj R2, Pg 25
AP/31

Student References Provided

APE059 AA1.01 - Accidental Liquid Radioactive-Waste Release

Ability to operate and / or monitor the following as they apply to the Accidental Liquid Radwaste Release: (CFR 41.7 / 45.5 / 45.6)

Radioactive-liquid monitor ....

401-9 Comments:

Remarks/Status

New KA

ILT48 ONS SRO NRC Examination QUESTION 25 25 **D**

APE060 AK3.03 - Accidental Gaseous-Waste Release

Knowledge of the reasons for the following responses as they apply to the Accidental Gaseous Radwaste: (CFR 41.5,41.10 / 45.6 / 45.13)

Actions contained in EOP for accidental gaseous-waste release ....

Given the following plant conditions:

- B GWD Tank rupture occurs
- 1RIA-32 (Aux Bldg Gas) in HIGH alarm
- 1RIA-39 (Cntl Rm Gas) in HIGH alarm
- Unit 1 AP/18 (Abnormal Release of Radioactivity) has been entered

1) AP/18 directs starting __(1)__Outside Air Booster Fan(s).

2) The Outside Air Booster Fan(s) __(2)__ utilize Unit 2 Main Purge Filters.

Which ONE of the following completes the statements above?

A. 1. ONLY one
   2. will

B. 1. two
   2. will

C. 1. ONLY one
   2. will NOT

D. 1. two
   2. will NOT

General Discussion

Answer A Discussion

Incorrect.
1st part is incorrect. Plausible since following a radiological accident in the Spent Fuel Pool, only one of the two Spent Fuel Filtered Exhaust Fans are started in accordance with AP/9 (Spent Fuel Damage). Also plausible to believe that only one fan would be started since the design basis is for the Outside Air Booster Fans is the most limiting LOCA fission product release and not a GWD Tank rupture.

2nd part is incorrect because the Booster Fans have their own intake ductwork including filters. It is plausible because other equipment, such as Spent Fuel Filtered Exhaust does utilize the RB Main Purge filters.

Answer B Discussion

Incorrect.
1st part is correct. With 1RIA-39 in alarm, AP/18 directs starting both Outside Air Booster Fans.

2nd part is incorrect because the Booster Fans have their own intake ductwork including filters. It is plausible because other equipment, such as Spent Fuel Filtered Exhaust does utilize the RB Main Purge filters.

Answer C Discussion

Incorrect.
1st part is incorrect. Plausible since following a radiological accident in the Spent Fuel Pool, only one of the two Spent Fuel Filtered Exhaust Fans are started in accordance with AP/9 (Spent Fuel Damage). Also plausible to believe that only one fan would be started since the design basis is for the Outside Air Booster Fans is the most limiting LOCA fission product release.
Second part is correct.

Answer D Discussion

1st part is correct. With 1RIA-39 in alarm, AP/18 directs starting both Outside Air Booster Fans.
2nd part is correct. The Outside Air Booster fans have their own intake duct work and filter setup.

Basis for meeting the KA

The question matches the KA by requiring knowledge of the actions contained in the AP for an accidental gas release..

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

RAD-RIA Obj R16
AP/18
1SA8 B-9

Student References Provided

APE060 AK3.03 - Accidental Gaseous-Waste Release
Knowledge of the reasons for the following responses as they apply to the Accidental Gaseous Radwaste: (CFR 41.5,41.10 / 45.6 / 45.13)
Actions contained in EOP for accidental gaseous-waste release ....

401-9 Comments:

Remarks/Status

Preview

BWA03 AA2.2 - Loss of NNI-Y
Ability to determine and interpret the following as they apply to the (Loss of NNI-Y)
(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:
Time = 0800:

- Reactor power = 25%
- 1A Main Feedwater Pump is operating
- 1SA-2/B-11 (ICS AUTO POWER FAILURE) actuated
- AP/23 (Loss of ICS Power) initiated

Time = 0805:

- Reactor trips

AP/23 (Loss of ICS Power)...

1) __(1)__ require tripping the 1A Main Feedwater Pump.

2) directs using __(2)__ to control decay heat removal.

Which ONE of the following completes the statements above?

A. 1. does NOT
2. ADVs

B. 1. does
2. ADVs

C. 1. does NOT
2. TBVs

D. 1. does
2. TBVs

General Discussion

Answer A Discussion

1st part is incorrect because AP/23 has an IAAT step that directs tripping all MFW pumps if the reactor trips. It is plausible because the MFW pump will trip on its own if a loss of ICS AUTO AND HAND power occurs but in this case, it doesn't automatically trip.

2nd part is incorrect because the TBVs will be available for hea removal. It is plausible because if it were a loss of ICS AUTO AND HAND power, it would be correct.

Answer B Discussion

1st part is correct. In AP/23, section 4B (Loss of ICS AUTO Power Only) step 2, IAAT a Rx trip occurs and ICS AUTO power is unavailable, THEN trip both MFDWPs.

2nd part is incorrect because the TBVs will be available for hea removal. It is plausible because if it were a loss of ICS AUTO AND HAND power, it would be correct.

Answer C Discussion

1st part is incorrect because AP/23 has an IAAT step that directs tripping all MFW pumps if the reactor trips. It is plausible because the MFW pump will trip on its own if a loss of ICS AUTO AND HAND power occurs but in this case, it doesn't automatically trip.

2nd part is correct. With ICS HAND power available, the TBVs are available for decay heat removal.

Answer D Discussion

1st part is correct. In AP/23, section 4B (Loss of ICS AUTO Power Only) step 2, IAAT a Rx trip occurs and ICS AUTO power is unavailable, THEN trip both MFDWPs.

2nd part is correct. With ICS HAND power available, the TBVs are available for decay heat removal.

Basis for meeting the KA

Requires knowledge of guidance provided in AP/23 on operating equipment with a loss of ICS power (NNI-Y).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	ILT44 Q24

Development References

ILT44 Q24
EAP-APG Obj R9
AP23

Student References Provided

BWA03 AA2.2 - Loss of NNI-Y

Ability to determine and interpret the following as they apply to the (Loss of NNI-Y)
(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

ILT48 ONS SRO NRC Examination QUESTION 27 27 **B**

BWA05 AK2.2 - Emergency Diesel Actuation

Knowledge of the interrelations between the (Emergency Diesel Actuation) 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 = 0800

- Reactor trip
- SBLOCA
- ES 1-6 actuate
- CT-1 Lockout

1) At Time = 0801, Main FDW Pumps __(1)__ automatically tripped.

2) In accordance with Rule 2 (Loss of SCM), an <u>initial</u> EFDW flow rate of __(2)__ gpm per SG is established to remove decay heat.

A. 1. have
 2. 300

B. 1. have NOT
 2. 300

C. 1. have
 2. 450

D. 1. have NOT
 2. 450

General Discussion

Answer A Discussion

Incorrect. First part is incorrect. At Time = 0800 the hotwell and condensate booster pumps, which provide suction to the Main FDW Pumps, trip with the loss of power. When the Main FDW Pump suction pressure is </= 235 psig, a 90 second timer starts. If the condition persists for 90 seconds, then the associated FDW Pump will trip. With ES actuated, power is restored through the Keowee Underground powerpath in approx. 15 seconds (12 - 18 sec). When ES actuates, the Keowee Units receive an emergency start signal. One second later a Load Shed signal is generated, which starts a 10 second timer for Transfer to Standby for a total of 11 seconds. At the 15 second mark, the Keowee unit will be up to speed and power will be restored, although the hotwell and condensate booster pumps do not auto restart.
Second part is correct. Rule 2 initially establishes 300 gpm EFDW flow per SG.

Answer B Discussion

Correct:
At Time = 0800 the hotwell and condensate booster pumps, which provide suction to the Main FDW Pumps, trip with the loss of power. When the Main FDW Pump suction pressure is </= 235 psig, a 90 second timer starts. If the condition persists for 90 seconds, then the associated FDW Pump will trip. The time given is at the 60 second mark and therefore, the Main FDW Pumps have not tripped.
Rule 2 initially establishes 300 gpm EFDW flow per SG. If flow can only be established to 1 SG, then 450 gpm is established to that SG.

Answer C Discussion

Incorrect. First part is incorrect. At Time = 0800 the hotwell and condensate booster pumps, which provide suction to the Main FDW Pumps, trip with the loss of power. When the Main FDW Pump suction pressure is </= 235 psig, a 90 second timer starts. If the condition persists for 90 seconds, then the associated FDW Pump will trip.
Second part is incorrect. Plausible since Rule 2 directs the operator to establish 450 gpm EFDW flow to the intact SG (if both SGs are not intact) when feeding to the LOSCM setpoint.

Answer D Discussion

Incorrect.
First part is correct.
Second part is incorrect. Plausible since Rule 2 directs the operator to establish 450 gpm EFDW flow to the intact SG (if both SGs are not intact) when feeding to the LOSCM setpoint

Basis for meeting the KA

Oconee's Emergency Diesel is the Keowee Hydro Units. The question requires knowledge of the interrelations between Keowee and the Motor Driven EFDW Pumps and the required EFDW flow for the SBLOCA event.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

EL-PSL Obj. 11
CF-FDW Obj. 06
EAP LOSCM Obj. 19
Rule 2

Student References Provided

BWA05 AK2.2 - Emergency Diesel Actuation
Knowledge of the interrelations between the (Emergency Diesel Actuation) 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

SYS003 K2.02 - Reactor Coolant Pump System (RCPS)
Knowledge of bus power supplies to the following: (CFR: 41.7)
CCW pumps ....

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1) The normal power supply to the 1A CC pump is __(1)__.

2) If MFBMP circuitry is activated (power to the Main Feeder Busses is lost) and then power is restored, CC pumps with their control switches in AUTO will start __(2)__.

Which ONE of the following completes the statements above?

A. 1. 1XL
2. immediately

B. 1. 1XL
2. after a 20 second time delay

C. 1. 1XS1
2. immediately

D. 1. 1XS1
2. after a 20 second time delay

General Discussion

Answer A Discussion

1st part is correct. 1XL is the normal power supply for the 1A CC pump.

2nd part is correct. With its switch in AUTO, after a MFBMP activation occurs, the CC pumps will receive a start signal. When power is restored, the pump will start.

Answer B Discussion

1st part is correct. 1XL is the normal power supply for the 1A CC pump.

2nd part is incorrect because the CC pump will start as soon as power is restored. It is plausible because there a 20 second time delay associated with the MFBMP logic. The MFBs have to be de-energized for 20 seconds before the MFBMP actuates.

Answer C Discussion

1st part is incorrect because the power supply is 1XL. It is plausible since 1XS1 is a motor control center that does supply major components including component cooling valve 1CC-7 however it does not supply power to the CC pumps. Second part is correct.

2nd part is correct. With its switch in AUTO, after a MFBMP activation occurs, the CC pumps will receive a start signal. When power is restored, the pump will start.

Answer D Discussion

1st part is incorrect because the power supply is 1XL. It is plausible since 1XS1 is a motor control center that does supply major components including component cooling valve 1CC-7 however it does not supply power to the CC pumps. Second part is correct.

2nd part is incorrect because the CC pump will start as soon as power is restored. It is plausible because there a 20 second time delay associated with the MFBMP logic. The MFBs have to be de-energized for 20 seconds before the MFBMP actuates.

Basis for meeting the KA

Requires knowledge of the 1A CC pump normal and emergency backup power supplies.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	MODIFIED	2010A Q35

Development References

2010A Q35
IC-ES Obj. 28
PNS-CC Obj R15

Student References Provided

SYS003 K2.02 - Reactor Coolant Pump System (RCPS)
Knowledge of bus power supplies to the following: (CFR: 41.7)
CCW pumps ....

401-9 Comments:

Remarks/Status

SYS004 K4.04 - Chemical and Volume Control System

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

Manual/automatic transfers of control ....

Given the following Unit 1 conditions:

0800:

- Reactor power = 100%
- MSLB inside containment occurs
- RCS pressure = 1575 psig

0805:

- RCS pressure = 1800 psig increasing
- The BOP is performing Encl 5.1 (ES Actuation)

1) At 0800, Diverse HPI __(1)__ actuated.

2) ES Channels 1 & 2 MANUAL Pushbuttons __(2)__ have to be depressed prior to throttling HPI.

Which ONE of the following completes the statements above?

A. 1) has
2) do

B. 1) has
2) do NOT

C. 1) has NOT
2) do

D. 1) has NOT
2) do NOT

General Discussion

Answer A Discussion

Incorrect.
1st part is incorrect because the setpoint for Diverse HPI = 1550 psig. It is plausible because if pressure had decreased to below 1550 psig, it would be correct.
2nd part is correct.

Answer B Discussion

Incorrect.
1st part is incorrect because the setpoint for Diverse HPI = 1550 psig. It is plausible because if pressure had decreased to below 1550 psig, it would be correct.

2nd part is incorrect. Plausible because if it were the RESET P/Bs, it would be correct.

Answer C Discussion

Correct.
Diverse HPI setpoint = 1550 psig.
ES Channels 1 and 2 have to be placed in manual (manual P/Bs depressed) in order to throttle HPI.

Answer D Discussion

Incorrect.
1st part is correct. Diverse HPI setpoint = 1550 psig.

2nd part is incorrect. Plausible because if it were the RESET P/Bs, it would be correct.

Basis for meeting the KA

The question requires knowledge of the ES design features that allow for transfering from automatic to manual control.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

IC-ES Obj 21
Encl 5.1
Encl 5.41

Student References Provided

SYS004 K4.04 - Chemical and Volume Control System
Knowledge of CVCS design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)
Manual/automatic transfers of control ....

401-9 Comments:

Remarks/Status

SYS004 K6.20 - Chemical and Volume Control System
Knowledge of the effect of a loss or malfunction on the following CVCS components: (CFR: 41.7 / 45.7)
Function of demineralizer, including boron loading and temperature limits

---

Given the following Unit 3 conditions:

- Reactor Power = 100%
- 3A Purification IX in service
- Letdown temperature <u>increases</u> by 5°F

1) RCS boron concentration will __(1)__ .

2) With control rods responding, assuming no other operator actions, taking ONLY the ICS Reactor Bailey station to HAND __(2)__ stop the rod motion.

Which ONE of the following completes the statements above?

A. 1. increase
2. will

B. 1. increase
2. will NOT

C. 1. decrease
2. will

D. 1. decrease
2. will NOT

General Discussion

Answer A Discussion

Incorrect.
First part is correct. Changing letdown temperature affects reactivity management due to the temperature effect on demineralizer resin. Increasing letdown temperature increases RCS boron and decreasing letdown temperature decreases RCS boron.

2nd part is plausible since the name itself implies that it controls signals to the reactor. Additionally, it does block changes in demand from getting the control rods so it would be plausible to believe that neutron error would also be blocked.

Answer B Discussion

CORRECT.
Changing letdown temperature affects reactivity management due to the temperature effect on demineralizer resin. Increasing letdown temperature increases RCS boron and decreasing letdown temperature decreases RCS boron.
Since neutron error drives control rod motion and neutron error is generated downstream of the Reactor Bailey, taking only the Rx bailey to hand will not stop the rod motion. It would freeze Rx demand but the boron would still impact neutron production therefore neutron error would still react and Control Rods would still respond.

Answer C Discussion

Incorrect. First part is incorrect. Plausible since it would be correct if letdown temperature decreased.
Second part is incorrect. Plausible since the name itself implies that it controls signals to the reactor. Additionally, it does block changes in demand from getting the control rods so it would be plausible to believe that neutron error would also be blocked.

Answer D Discussion

Incorrect. First part is incorrect. Plausible since it would be correct if letdown temperature decreased.
Second part is correct. Since neutron error drives control rod motion and neutron error is generated downstream of the Reactor Bailey, taking only the Rx bailey to hand will not stop the rod motion. It would freeze Rx demand but the boron would still impact neutron production therefore neutron error would still react and Control Rods would still respond.

Basis for meeting the KA

Requires knowledge of how a malfunction of a demineralizer (boron loading) will affect the rest of the plant. In this case the "malfunction" would be that procedurally the demineralizer should have been saturated to current RCS boron before being placed in service. An incorrectly boron saturated demineralizer being placed in service is effectively a malfunction.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	ILT46 Q30

Development References

ILT46 Q30
PNS-HPI Obj 07
ICS picture

Student References Provided

SYS004 K6.20 - Chemical and Volume Control System
Knowledge of the effect of a loss or malfunction on the following CVCS components: (CFR: 41.7 / 45.7)
Function of demineralizer, including boron loading and temperature limits

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 31 31 **C**

SYS005 K2.01 - Residual Heat Removal System (RHRS)
Knowledge of bus power supplies to the following: (CFR: 41.7)
RHR pumps ....

Given the following Unit 1 conditions:

- Reactor Power = 100%
- 1TE bus lockout occurs

Which ONE of the following contains ONLY pumps that are still being powered from 4160V switchgear.

A. 1B HPIP, 1B LPIP

B. 1B HPIP, 1C LPIP

C. 1C HPIP, 1B LPIP

D. 1C HPIP, 1C LPIP

ILT48 ONS SRO NRC Examination QUESTION 31 31 **C**

General Discussion

Answer A Discussion
Incorrect. Plausible since both pumps are from the "B" trains.

Answer B Discussion
Incorrect. Plausible since one pump is from "B" train and the other from "C" train

Answer C Discussion
Correct. Neither 1C HPIP nor 1B LPIP are powered from 1TE.

Answer D Discussion
Incorrect. Plausible since ONS component power does not strictly adhere to the "train" principle of alignment.

Basis for meeting the KA
Requires knowledge of power supplies to LPI pumps

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	MODIFIED	ILT46 Q31

Development References
ILT46 Q31
IC-ES Obj R20
Power Supply chart

Student References Provided

SYS005 K2.01 - Residual Heat Removal System (RHRS)
Knowledge of bus power supplies to the following: (CFR: 41.7)
RHR pumps ....

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 32 32 **C**

SYS006 K6.10 - Emergency Core Cooling System (ECCS)
Knowledge of the effect of a loss or malfunction on the following will have on the ECCS: (CFR: 41.7 / 45.7)
Valves ....

Given the following Unit 1 conditions:
Time = 0400

- Reactor power = 100%

Time = 0405

- RCS pressure = 550 psig decreasing

Time = 0445

- RCS pressure = 200 psig decreasing

Time = 0450

- RCS pressure = 100 psig decreasing
- 1LP-17 is failed CLOSED

1) In accordance with OMP 1-18 Attachment A (Licensed Operator Memory Items) at Time = 0405, the LATEST time the LPI pumps are required to be secured is __(1)__.

2) At Time = 0450, LPI flow __(2)__ entering the core through BOTH LPI injection nozzles.

Which ONE of the following completes the statements above?

A. 1. 0425
2. is

B. 1. 0425
2. is NOT

C. 1. 0435
2. is

D. 1. 0435
2. is NOT

General Discussion

Answer A Discussion

Incorrect. First part is plausible in the candidate has the misconception that the time allowed is 20 minutes. 20 minutes is the time for several time critical tasks including "starting the SSF RCMUP during a HELB" and "transferring MDEFDWP suction to the hotwell" .
Second part is correct.

Answer B Discussion

Incorrect. First part is plausible if the candidate has the misconception that the time allowed is 20 minutes. 20 minutes is the time for several time critical tasks including "starting the SSF RCMUP during a HELB" and "transferring MDEFDWP suction to the hotwell" .
Second part is plausible because the main injection valve for the "A" header (1LP-17) is closed therefore without knowledge of the LPI crossover alignment downstream of 1LP-17 and 1LP-18 the candidate would make this choice.

Answer C Discussion

Correct. Per OMP 1-18 Attachment A, the LPI pumps must not operate against a shutoff head greater than 30 minutes. The LPI pumps would be manually restarted as RCS pressure decreases below 200 psig. Due to the crossover mod LPI flow will enter the core through both nozzles even with 1LP-17 closed since the LPI crossover mod connected the two LPI headers downstream of LP=17 and LP-18.

Answer D Discussion

Incorrect. First part is correct..
Second part is plausible because the main injection valve for the "A" header (1LP-17) is closed therefore without knowledge of the LPI crossover alighnment downstream of 1LP-17 and 1LP-18 the candidate would make this choice.

Basis for meeting the KA

Question requires knowledege of ECCS response when a valve is failed closed. 1LP-17 is an ECCS valve in the 1A LPI header.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT41 Q32

Development References

ILT41 Q32
EAP-ESA Obj R12, R17
OMP 1-18

Student References Provided

SYS006 K6.10 - Emergency Core Cooling System (ECCS)
Knowledge of the effect of a loss or malfunction on the following will have on the ECCS: (CFR: 41.7 / 45.7)
Valves ....

401-9 Comments:

Remarks/Status

SYS007 A1.01 - Pressurizer Relief Tank/Quench Tank System (PRTS)

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

Maintaining quench tank water level within limits ....

Given the following Unit 2 conditions:

- Reactor power = 100%
- Quench Tank is being pumped to 2A BHUT using the Quench Tank Pump AND the Component Drain Pump

1) In accordance with OP/2/A/1104/017 (Quench Tank Operations), Quench Tank Level shall be maintained at a MAXIMUM of __(1)__ inches.

2) If ES -1/2 actuates, this flow path __(2)__ automatically isolate.

Which ONE of the following completes the statements above?

A. 1. 90
2. will

B. 1. 90
2. will NOT

C. 1. 100
2. will

D. 1. 100
2. will NOT

General Discussion

Answer A Discussion

1st part is correct. IAW OP/1104/017 limits and precautions, QT level shall be maintained between 80 and 90 inches.

2nd part is correct. Upon receiving an ES signal, 2CS-5 (ES-1) and 2CS-6 (ES-2) will close and isolate the flow path.

Answer B Discussion

1st part is correct. IAW OP/1104/017 limits and precautions, QT level shall be maintained between 80 and 90 inches.

2nd part is incorrect because if you are pumping the QT to the 2A BHUT (OP/2/A/1104/017 Encl 4.1) 2CS-5 and 2CS-6 will close upon receiving an ES 1 & 2 signal. It is plausible because if you were pumping from the 2A BHUT TO the Quench Tank (OP/2/A/1104/017 Encl 4.2), it would be correct because this flow path is performed with manual valves that will not isolate upon receiving an ES 1/2 signal.

Answer C Discussion

1st part is incorrect because Unit 2 upper limit is 90 inches. It is plausible because if it were Unit 1, it would be correct.

2nd part is correct. Upon receiving an ES signal, 2CS-5 (ES-1) and 2CS-6 (ES-2) will close and isolate the flow path.

Answer D Discussion

1st part is incorrect because Unit 2 upper limit is 90 inches. It is plausible because if it were Unit 1, it would be correct.

2nd part is incorrect because if you are pumping the QT to the 2A BHUT (OP/2/A/1104/017 Encl 4.1) 2CS-5 and 2CS-6 will close upon receiving an ES 1 & 2 signal. It is plausible because if you were pumping from the 2A BHUT TO the Quench Tank (OP/2/A/1104/017 Encl 4.2), it would be correct because this flow path is performed with manual valves that will not isolate upon receiving an ES 1/2 signal.

Basis for meeting the KA

Requires knowledge of Quench Tank level limits associated with the system.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	MODIFIED	ILT46 Q34

Development References

ILT46 Q34
PNS-CS Obj 10
1104/17

Student References Provided

SYS007 A1.01 - Pressurizer Relief Tank/Quench Tank System (PRTS)

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

Maintaining quench tank water level within limits ....

401-9 Comments:

Remarks/Status

SYS008 A4.01 - Component Cooling Water System (CCWS)
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5)
CCW indications and controls ....

Given the following Unit 1 conditions:
Time = 0400:

- Reactor power = 100%
- Component Cooling Return Flow = 563 gpm decreasing
- 1SA-09/C-1 (Component Cooling Return Flow Low) actuates

Time = 0402

- Component Cooling Return Flow = 103 gpm decreasing
- The Standby CC pump has NOT started
- CC Surge Tank level = 18 inches stable

1) At Time = 0400, Statalarm 1SA-09/C-1 __(1)__ valid.

2) At Time = 0402, in accordance with 1SA-09/C-1 ARG, the Standby CC pump __(2)__ be manually started.

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

ILT48 ONS SRO NRC Examination QUESTION 34 34 **A**

General Discussion

Answer A Discussion

Correct. 1SA-09/C-1 set point is 575 gpm. The Standby CC pump did not start and should be started after verifying that CC surge tank level is > 12 inches.

Answer B Discussion

Incorrect. First part is correct. Second part is plausible because it would be correct if CC level were less than 12 inches.

Answer C Discussion

Incorrect. First part is plasuible because it is above the setpoint for Statalarm 1SA-09/B-1 (CRD Return Flow Low) of 138 gpm. Second part is correct.

Answer D Discussion

Incorrect. First part is plasuible because it is above the setpoint for Statalarm 1SA-09/B-1 (CRD Return Flow Low) of 138 gpm. Second part is plausible because it would be correct if CC level were less than 12 inches.

Basis for meeting the KA

Question requires knowledge of Component Cooling related indications and controls it the control room.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT42 Q7

Development References

ILT42 Q7
PNS-CC Obj 08
ARG

Student References Provided

SYS008 A4.01 - Component Cooling Water System (CCWS)
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5)
CCW indications and controls ....

401-9 Comments:

Remarks/Status

SYS010 K3.01 - Pressurizer Pressure Control System (PZR PCS)
Knowledge of the effect that a loss or malfunction of the PZR PCS will have on the following: (CFR: 41.7 / 45.6)
RCS ....

Given the following Unit 1 conditions:
Initial conditions:

- Reactor power = 100%
- 1RC-1 failed OPEN
- RCS pressure = 2150 psig decreasing
- AP/44 (Abnormal Pressurizer Pressure Control) initiated

Current conditions:

- 1RC-3 failed OPEN

1) Pressurizer heaters __(1)__ be able to maintain RCS pressure above the RPS trip setpoint.

2) AP/44 will initially direct stopping the __(2)__ to help mitigate the failure.

Which ONE of the following completes the statements above?

A. 1. will NOT
2. 1A1 RCP ONLY

B. 1. will NOT
2. 1A1 and 1A2 RCPs

C. 1. will
2. 1A1 RCP ONLY

D. 1. will
2. 1A1 and 1A2 RCPs

General Discussion

Answer A Discussion

Incorrect. First part is correct. Second part is plausible because the 1A1 RCP is the "spray" pump.

Answer B Discussion

Correct. Heaters will not maintain pressure with the Spray valve open. Pressure will continue to decrease. AP/44 will initially stop the 1A1 and 1A2 RCPs.

Answer C Discussion

Incorrect. First part is plausible because the candidate may have the misconception that Pzr heaters could overcome Pzr spray. Second part is plausible because the 1A1 RCP is the "spray" pump.

Answer D Discussion

Incorrect. First part is plausible because the candidate may have the misconception that Pzr heaters could overcome Pzr spray. Second part is correct.

Basis for meeting the KA

Question requires knowledge of how a failure of Pressurizer pressure control system (PZR spray) would affect RCS pressure.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT45 Q37

Development References

ILT45 Q37
PNS-PZR Obj. 11
EAP-APG Obj. R9
AP/44

Student References Provided

SYS010 K3.01 - Pressurizer Pressure Control System (PZR PCS)
Knowledge of the effect that a loss or malfunction of the PZR PCS will have on the following: (CFR: 41.7 / 45.6)
RCS ....

401-9 Comments:

Remarks/Status

SYS012 K5.01 - Reactor Protection System (RPS)

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

DNB ....

The DNBR Safety Limit is applicable in MODE(s) __(1)__ and the High flux RPS trip setpoint __(2)__ designed to prevent exceeding that limit.

Which ONE of the following completes the statement above?

A. 1. one ONLY
2. is

B. 1. one ONLY
2. is NOT

C. 1. one AND two
2. is

D. 1. one AND two
2. is NOT

General Discussion

Answer A Discussion

Incorrect. First part is incorrect but plausible since the Safety Limits do apply in MODE 1. Also it would be plausible to question the logic of a DNBR limit being in place in MODE 2 since power is limited to 5% in that MODE.
Second part is correct.

Answer B Discussion

Incorrect. First part is incorrect but plausible since the Safety Limits do apply in MODE 1. Also it would be plausible to question the logic of a DNBR limit being in place in MODE 2 since power is limited to 5% in that MODE.
Second part is plausible since the High Flux trip is also designed to protect the fuel centerline temperature Safety Limit therefore under the misconception that a trip only protects against one safety limit it would be more logical to choose the High Temperature trip as the DNBR protection and high flux as the fuel centerline melt protection.

Answer C Discussion

Correct. The DNBR and fuel centerline temperature Safety Limits are applicable in both Mode 1 and 2. The High flux RPS trip is designed to protect against exceeding both the DNBR and Fuel Centerline temperature Safety Limits.

Answer D Discussion

Incorrect. First part is correct.
Second part is plausible since the High Flux trip is also designed to protect the fuel centerline temperature Safety Limit therefore under the misconception that a trip only protects against one safety limit it would be more logical to choose the High Temperature trip as the DNBR protection and high flux as the fuel centerline melt protection.

Basis for meeting the KA

Question requires knowledge of the operational implications of exceeding RPS trip setpoints as they relate to DNBR.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	MODIFIED	ILT42 Q38

Development References

ILT42 Q38
IC-RPS Obj R4

Student References Provided

SYS012 K5.01 - Reactor Protection System (RPS)
Knowledge of the operational implications of the following concepts as the apply to the RPS: (CFR: 41.5 / 45.7)
DNB ....

401-9 Comments:

Remarks/Status

SYS013 K4.12 - Engineered Safety Features Actuation System (ESFAS)
Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the following : (CFR: 41.7)
Safety injection block ....

Given the following Unit 1 conditions:

- RCS cooldown in progress
- RCS pressure = 1730 psig slowly decreasing

1) 1SA-7/D-6 (ES HPI Bypass Permit) __(1)__ actuated.

2) Once bypassed, HPI ES __(2)__ AUTOMATICALLY reinstate when RCS pressure is returned to normal operating pressure.

Which ONE of the following completes the statements above?

A. 1. is
   2. does

B. 1. is
   2. does NOT

C. 1. is NOT
   2. does

D. 1. is NOT
   2. does NOT

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C

General Discussion

Answer A Discussion

Incorrect. First part is incorrect. Plausible since pressure is below 1740 psig which is the pressure where it is reinstated. Second part is correct.

Answer B Discussion

Incorrect. First part is incorrect. Plausible since pressure is below 1740 psig which is the pressure where it is reinstated. Second part is incorrect. Plausible since other safety systems like AFIS do not auto reinstate once actuated.

Answer C Discussion

CORRECT. 1SA-7/D6 (ES HPI Bypass Permit) actuates at 1715 psig decreasing RCS pressure. Once actuated, it will reset at 1740 psig RCS pressure increasing. Therefore with RCS pressure 1730 psig decreasing , the alarm is not actuated.
If HPI ES is bypassed, it will auto reinstate when RCS pressure increases above 1740 psig.

Answer D Discussion

Incorrect. First part is correct. Second part is incorrect. Plausible since other safety systems like AFIS do not auto reinstate once actuated.

Basis for meeting the KA

Question requires knowledge of design features and interlock operation of the HPI portion of ES as it relates to blocking actuation.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT45 Q32

Development References

ILT45 Q32
EC-ES Obj 05

Student References Provided

SYS013 K4.12 - Engineered Safety Features Actuation System (ESFAS)

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

Safety injection block ....

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 38 38 **D**

SYS022 A3.01 - Containment Cooling System (CCS)
Ability to monitor automatic operation of the CCS, including: (CFR: 41.7 / 45.5)
Initiation of safeguards mode of operation ....

Given the following Unit 1 Conditions:
Time = 0500:

- Reactor power = 25% stable
- 1A and 1C RBCUs operating in HIGH speed
- 1B RBCU is operable and OFF

Time = 0501:

- A LOCA occurs
- ES channels 1-5 actuate
- ES channel 6 fails to actuate
- A LOOP occurs

Time = 0505:

- Offsite power is restored to Unit 1

At Time = 0506, 1C RBCU is __(1)__ and 1B RBCU is __(2)__.

Which ONE of the following completes the statement above?

A. 1. operating in LOW speed
2. operating in LOW speed

B. 1. operating in LOW speed
2. OFF

C. 1. OFF
2. operating in LOW speed

D. 1. OFF
2. OFF

ILT48 ONS SRO NRC Examination QUESTION 38

38

D

General Discussion

Answer A Discussion

INCORRECT. First part is plausible if the applicant misapplies the 3-minute time delay AND confuses which RBCU restarts from ES-5. (A RBCU restarts from ES-5 and C RBCU restarts from ES-6). Second part is plausible if the applicant misapplies the 3-minute time delay and starts the clock at ES actuation rather than restoration of power.

Answer B Discussion

INCORRECT. First part is plausible if the applicant misapplies the 3-minute time delay AND confuses which RBCU restarts from ES-5. (A RBCU restarts from ES-5 and C RBCU restarts from ES-6). Second part is correct.

Answer C Discussion

INCORRECT. First part is correct. Second part is plausible if the applicant misapplies the 3-minute time delay and starts the clock at ES actuation rather than restoration of power.

Answer D Discussion

CORRECT. In the case of a simultaneous LOCA and LOOP, the 3-minute time delay starts when power is restored (rather than when ES actuates). Therefore only 1 minute has passed since power was restored.

Basis for meeting the KA

The question requires the applicant to know that ES-5/6 actuate on high containment pressure, and then determine the RBCU configuration as a result of the actuation, including the 3 minute time delay. The mixed speed circuit stops all running RBCUs on receipt of ES Ch-5/6 actuation, and after a 3-minute time delay, restarts them in LOW. In the case of a simultaneous LOCA and LOOP, the 3-minute time delay starts when power is restored (rather than when ES actuates).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT43 Q39

Development References

ILT43 Q39
PNS-RBC Obj. 01

Student References Provided

SYS022 A3.01 - Containment Cooling System (CCS)

Ability to monitor automatic operation of the CCS, including: (CFR: 41.7 / 45.5)

Initiation of safeguards mode of operation ....

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination **QUESTION 39** 39 **D**

SYS026 A1.05 - Containment Spray System (CSS)

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

Chemical additive tank level and concentration ....

Following a LBLOCA, __(1)__ is added to the Reactor Building Emergency Sump to __(2)__.

Which ONE of the following completes the statement above?

A. 1. LIOH (Lithium Hydroxide)
 2. minimize Hydrogen production from the Boric Acid reaction with Zircoloy

B. 1. LIOH (Lithium Hydroxide)
 2. aid in keeping Iodine in solution, ultimately reducing offsite dose

C. 1. TSP (Trisodium Phosphate Dodecahydrate)
 2. minimize Hydrogen production from the Boric Acid reaction with Zircoloy

D. 1. TSP (Trisodium Phosphate Dodecahydrate)
 2. aid in keeping Iodine in solution, ultimately reducing offsite dose

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General Discussion

Answer A Discussion

Incorrect. First part is plausible because TSP is added to the Emergency sump to control PH and LIOH is added to RCS to control PH. Second part is plausible since it does inhibit H2 production due to the boric acid reaction with Zinc and Aluminum.

Answer B Discussion

Incorrect. First part is plausible because TSP is added to the Emergency sump to control PH and LIOH is added to RCS to control PH. Second part is correct.

Answer C Discussion

First part is correct. Second part is plausible since it does inhibit H2 production due to the boric acid reaction with Zinc and Aluminum.

Answer D Discussion

Correct. TSP is added to the Emergency Sump by way of TSP baskets as containment water level increases following a LOCA. The TSP is added to control PH which controls the amount of Iodine that comes out of solution which reduces offsite doses.

Basis for meeting the KA

CE stated that asking question regarding the TSP baskets could be used to match this KA. Requires the ability to monitor changes in parameters (offsite doses) based on addition of TSP. The TSP is added to containment water which is utilized by the RBS system.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT43 Q40

Development References

ILT43 Q40
PNS-RBS Obj 1

Student References Provided

SYS026 A1.05 - Containment Spray System (CSS)

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

Chemical additive tank level and concentration ....

401-9 Comments:

Remarks/Status

SYS026 K1.01 - Containment Spray System (CSS)

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

ECCS ....

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1) The actual SETPOINT for Reactor Building Spray actuation is __(1)__ psig.

2) The 1BS-2 position if ES Channel 8 fails to actuate is __(2)__.

Which ONE of the following completes the statements above?

A. 1. 10
2. Open

B. 1. 10
2. Closed

C. 1. 15
2. Open

D. 1. 15
2. Closed

ILT48 ONS SRO NRC Examination QUESTION 40

40

B

General Discussion

Answer A Discussion

Incorrect: First part is correct. Second part is plausbile since there are mulitiple ES valve components that are normally in their ES position. (Ex. HP-27 is C HPI pump disch valve and is normally open). Also plausble since the pump suction valves are MOV's and are normally open. Also plausble since we leave BS-15 (BS header drain valve located in RB) open to ensure we do not spray down containment at power.

Answer B Discussion

Correct: ES-7&8 actual setpoint is 10 psig and the RB Spray pump discharge valves (BS-1&2) are normally closed and are required to open on ES signal.

Answer C Discussion

Incorrect: First part is plausible since <15 psig is the TS required setpoint. Second part is plausbile since there are mulitiple ES valve components that are normally in their ES position. (Ex. HP-27 is C HPI pump disch valve and is normally open). Also plausble since the pump suction valves are MOV's and are normally open. Also plausble since we leave BS-15 (BS header drain valve located in RB) open to ensure we do not spray down containment at power.

Answer D Discussion

Incorrect: First part is plausible since <15 psig is the TS required setpoint. Second part is correct.

Basis for meeting the KA

Requires knowledge of the relationship between CSS (building spray) and ECCS (initiation).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT40 Q43

Development References

ILT40 Q43
PNS-BS Obj. 03 & 06

Student References Provided

SYS026 K1.01 - Containment Spray System (CSS)

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

ECCS ....

401-9 Comments:

Remarks/Status

SYS039 A4.03 - Main and Reheat Steam System (MRSS)
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
MFW pump turbines ....

Given the following plant conditions:
Time = 1200

- Unit 2 Reactor power = 75%
- Reactor Diamond, BOTH FDW masters, and the Main and Startup FDW valves are in HAND

Time = 1215

- Reactor trip occurs

1) At Time = 1200, the Unit 2 Main Feedwater Pump Turbines are being controlled by their associated __(1)__.

2) At Time = 1215, runback of Main Feedwater flow __(2)__ occur AUTOMATICALLY.

Which ONE of the following completes the statements above?

A. 1. Motor Gear Unit
   2. will

B. 1. Motor Gear Unit
   2. will NOT

C. 1. Motor Speed Changer
   2. will

D. 1. Motor Speed Changer
   2. will NOT

ILT48 ONS SRO NRC Examination QUESTION 41

41

A

General Discussion

Answer A Discussion

Correct.
1st part is correct because the Motor Gear Unit (MGU) is the normal speed control for the Main FDW Pumps. At 75% power, both Main FDW Pumps are operating and since there is no mention of the Hand Jack switch, the FDWPs are being controlled by the MGU. In order for the FDWP to be controlled by the Motor Speed Changer (MSC), the FDWP would have to be on the Hand Jack. The Hand Jack switch is used when it is necessary to remove the MGU from service for maintenance, etc. while the FDWP is operating. Placing the FDWP on Hand Jack simulates the MGU on the high speed stop. FDWP control is the lowest speed signal called for, which when on the Hand Jack, would be the MSC.
2nd part is correct because upon receiving a reactor trip signal, the MFWPTs, Main & Startup FDW valves will transfer to AUTO and process the runback signal.

Answer B Discussion

Incorrect.
1st part is correct.
2nd part is incorrect because the upon receiving a reactor trip signal, the MFWPTs, Main & Startup FDW valves will transfer to AUTO and process the runback signal. It is plausible becasuse if they did not receive a separate signal on a reactor trip, the runback signal from the ICS "Integrated Master" would not pass through the Loop Masters to runback feedwater and this answer would be correct.

Answer C Discussion

Incorrect.
1st part is incorrect. Plausible since it would be correct if the FDWPTs were on the Hand Jack.
2nd part is correct because upon receiving a reactor trip signal, the MFWPTs, Main & Startup FDW valves will transfer to AUTO and process the runback signal.

Answer D Discussion

Incorrect.
1st part is incorrect. Plausible since it would be correct if the FDWPTs were on the Hand Jack.
2nd part is incorrect because the upon receiving a reactor trip signal, the MFWPTs, Main & Startup FDW valves will transfer to AUTO and process the runback signal. It is plausible because if they did not receive a separate signal on a reactor trip, the runback signal from the ICS "Integrated Master" would not pass through the Loop Masters to runback feedwater and this answer would be correct.

Basis for meeting the KA

Requires the ability to monitor for proper steam supply to the main feedwater pumps. Reheat Steam is consistent with D bleed.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References

CF-FPT obj 02
STG-ICS

Student References Provided

SYS039 A4.03 - Main and Reheat Steam System (MRSS)
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
MFW pump turbines ....

401-9 Comments:

Remarks/Status

SYS039 2.1.32 - Main and Reheat Steam System (MRSS)
SYS039 GENERIC
Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)

In accordance with Limits and Precautions of OP/1/A/1106/014 (Moisture Separator Reheater), the MAXIMUM allowed side to side delta Temperature across the Low Pressure Turbines during normal operation is __(1)__ degrees F and we ensure this limit is not exceeded by throttling valves controlling the steam supply to the __(2)__ as power level changes.

Which ONE of the following completes the statement above?

A. 1. 50
2. First Stage Reheaters

B. 1. 50
2. Second Stage Reheaters

C. 1. 100
2. First Stage Reheaters

D. 1. 100
2. Second Stage Reheaters

ILT48 ONS SRO NRC Examination QUESTION 42

42

B

General Discussion

Answer A Discussion

Incorrect, First part is correct. Second part is plausible since this method would control side to side delta T's on the LP turbine however the this is not how it is controlled

Answer B Discussion

Correct. Per L&P's 50 degrees is the maximum side to side delta T allowed and this is controlled by throttling MS-112 and 173 with power level. These valves control the steam being supplied to the SSRH's and therefore the steam temperature being admitted to the LP turbines.

Answer C Discussion

Incorrect. First part is plausible since 100 degrees/hr is the maximum SSRH tube HUR allowed by L&P's.
Second part is plausible since this method would control side to side delta T's on the LP turbine however the this is not how it is controlled

Answer D Discussion

Incorrect. First part is plausible since 100 degrees/hr is the maximum SSRH tube HUR allowed by L&P's.
Second part is correct..

Basis for meeting the KA

Requires the knowledge of a system L&P in the MSR procedure and how we apply that limit during power maneuvering (how we control the temperature below the limit).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

STG-MSR obj 08
1106/14

Student References Provided

SYS039 2.1.32 - Main and Reheat Steam System (MRSS)
SYS039 GENERIC
Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)

401-9 Comments:

Remarks/Status

SYS059 A2.07 - Main Feedwater (MFW) System

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

Tripping of MFW pump turbine ....

Given the following Unit 1 conditions:

- Reactor power = 80% stable
- Reactor Diamond and both Feedwater masters are in HAND
- Feedwater Transient occurs which results in one of the Main Feedwater Pumps tripping on High Discharge Pressure

1) The Main Feedwater Pump with the LOWER Discharge Pressure trip setpoint is the __(1)__ Main Feedwater Pump.

2) In accordance with AP/1 (Unit Runback), a Manual power reduction to a MINIMUM of __(2)__ % power will be performed.

Which ONE of the following completes the statements above?

A. 1. 1A
2. 65

B. 1. 1A
2. 74

C. 1. 1B
2. 65

D. 1. 1B
2. 74

General Discussion

Answer A Discussion

Incorrect. First part is plausible since there are only 2 MFDWP's and since 1A is first alphabetically it would be logical to assume it would be the one you want to trip first if you did not know the specific setpoints.
Second part is correct.

Answer B Discussion

Incorrect. First part is plausible since there are only 2 MFDWP's and since 1A is first alphabetically it would be logical to assume it would be the one you want to trip first if you did not know the specific setpoints.
Second part is plausible since it would be correct for a RCP trip.

Answer C Discussion

Correct. The 1B MFDWP disch pressure setpoint is 1240 psig and the 1A MFDWP disch pressure trip setpoint is 1275 psig.

Answer D Discussion

Incorrect. First part is correct.
Second part is plausible since it would be correct for a RCP trip.

Basis for meeting the KA

Question requires the candidate to predict which FDW pump will trip first during a malfucntion that impacts FDWP discharge pressure and then use procedures to mitigate the consequences.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References	Student References Provided
CF-FPT Obj 14 AP/1	

SYS059 A2.07 - Main Feedwater (MFW) System

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

Tripping of MFW pump turbine ....

401-9 Comments:	Remarks/Status
	NEW KA

SYS059 A3.06 - Main Feedwater (MFW) System
Ability to monitor automatic operation of the MFW, including: (CFR: 41.7 / 45.5)
Feedwater isolation ....

Given the following Unit 1 conditions:
Initial Conditions:

- Reactor power = 100%
- Two AFIS pressure transmitters on 1A SG fail high

Current Conditions:

- 1A Main Steam Line break occurs
- 1A SG pressure = 480 psig rapidly decreasing

1) AFIS __(1)__ actuated.

2) Rule 5 (Main Steam Line Break) will direct the operator to __(2)__.

Which ONE of the following completes the statements above?

A. 1. is
2. open 1AS-40 while closing 1MS-47

B. 1. is
2. select OFF for 1A MD EFDW Pump

C. 1. is NOT
2. open 1AS-40 while closing 1MS-47

D. 1. is NOT
2. select OFF for 1A MD EFDW Pump

General Discussion

Answer A Discussion

Incorrect. First part is correct. Second part is plausible since it would be correct if the MSLB were in the B SG. With the leak in the A SG, Rule 5 will not direct transfer of the CSAE steam supply.

Answer B Discussion

Correct: AFIS uses 4 transmitters. Any two of the 4 decreasing to 550 psig will result in an AFIS actuation therefore AFIS will still automatically initiate as long as there are two PTs available. Rule 5 will direct placing the switch for the 1A MD EFDWP in OFF.

Answer C Discussion

Incorrect: AFIS will auto initiate. Plausible if AFIS Logic is assumed to be disabled with 2 pressure switches failed which is the case for numerous instrument strings (2/3 logic to mitigate a single failure). Second part is plausible since it would be correct if the MSLB were in the B SG. With the leak in the A SG, Rule 5 will not direct transfer of the CSAE steam supply.

Answer D Discussion

Incorrect: AFIS will auto initiate. Plausible if AFIS Logic is assumed to be disabled with 2 pressure switches failed which is the case for numerous instrument strings (2/3 logic to mitigate a single failure). Second part is correct. Additionally, second part is plausible if you believe that AFIS has not actuated since the MDEFWP is placed in OFF anytime there is a MSLB on its associated SG. This means that even if pressure does not
decrease to the AFIS setpoint you would still place the MDEFWP in OFF.

Basis for meeting the KA

Requires knowledge of design features and automatic operation of the Main Feedwater system as it relates to the Automatic Feedwater Isolation System.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	2009B NRC Exam Q43

Development References

2009B NRC Exam Q43
CF-FDW Obj. 19
EAP-EHT Obj. R14
EOP Rule 5

Student References Provided

SYS059 A3.06 - Main Feedwater (MFW) System
Ability to monitor automatic operation of the MFW, including: (CFR: 41.7 / 45.5)
Feedwater isolation ....

401-9 Comments:

Remarks/Status

SYS061 2.4.8 - Auxiliary / Emergency Feedwater (AFW) System
SYS061 GENERIC
Knowledge of how abnormal operating procedures are used in conjunction with EOPs. (CFR: 41.10 / 43.5 / 45.13)

Given the following Unit 1 conditions:

- Reactor power = 100%
- Confirmed flooding of the Turbine Building is in progress

1) The EOP TBF tab __(1)__ state that AP/10 (Turbine Building Flood) will be performed in parallel with the tab.

2) Actions taken in accordance with AP/10 will require that __(2)__ Feedwater be used for a long term source of water to the Steam Generators.

Which ONE of the following completes the statements above?

A. 1. does
2. Emergency

B. 1. does
2. Main

C. 1. does NOT
2. Emergency

D. 1. does NOT
2. Main

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45

A

General Discussion

Answer A Discussion

CORRECT. A note at the beginning of AP/10 and the EOP TBF tab states that the AP will be used in parallel with the EOP. The AP attempts to identify and stop the source of the water while the EOP focuses on core protection. AP/10 directs securing the CCW pumps in order to stop or reduce the in leakage. A note in the AP explains that this will cause a loss of vacuum and therefore a loss of the Main Feedwater pumps.

Answer B Discussion

Incorrect. First part is correct. In the case of AP/10, both the AP and the EOP TBF tab state that AP/10 is to be performed in parallel with the EOP TBF tab.
Second part is plausible since without considering the loss of Vacuum caused by the securing of CCW pumps directed in AP/10, Main FDW would be available for long term DHR as needed.

Answer C Discussion

Incorrect. First part is incorrect. Plausible since it would be correct for other Aps, which are not performed in parallel with the EOP.
Second part is correct.

Answer D Discussion

Incorrect. First part is incorrect. Plausible since it would be correct for other APs, which are not performed in parallel with the EOP.
Second part is plausible since without considering the loss of Vacuum caused by the securing of CCW pumps directed in AP/10, Main FDW would be available for long term DHR as needed.

Basis for meeting the KA

Required knowledge of how the EOP and AP/10 are used in conjunction with each other with consequences being EFDW being used as long term SG feedwater source.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References	Student References Provided
EAP-APG R9 AP/10	

SYS061 2.4.8 - Auxiliary / Emergency Feedwater (AFW) System
SYS061 GENERIC
Knowledge of how abnormal operating procedures are used in conjunction with EOPs. (CFR: 41.10 / 43.5 / 45.13)

401-9 Comments:	Remarks/Status

SYS062 K1.04 - AC Electrical Distribution System

Knowledge of the physical connections and/or cause-effect relationships between the ac distribution sys- tem and the following systems : (CFR: 41.2 to 41.9)

Off-site power sources ....

---

1) Automatic control circuits will close the associated feeder breakers of 1X7, 2X4 & 3X4 after a load shed has occurred and a __(1)__ second timer has timed out.

2) The reason 1X7, 2X4 & 3X4 load shed is to __(2)__.

Which ONE of the following completes the statements above?

A. 1. 30
2. ensure the integrity of the RCP seals

B. 1. 30
2. prevent overloading the CT-4 or CT-5 transformers

C. 1. 60
2. ensure the integrity of the RCP seals

D. 1. 60
2. prevent overloading the CT-4 or CT-5 transformers

General Discussion

Answer A Discussion

INCORRECT. First part is plausible because X5 & X6 load centers are load shed under certain conditions and the auto reclose time delay for these is 30 seconds. Second part is plausible because this is one of the purposes of the Main Feeder Bus Monitor Panels. MFBMP's assure the integrity of the RCP seals by insuring that seal injection and component cooling flows are regained following a loss of power.

Answer B Discussion

INCORRECT. First part is plausible because X5 & X6 load centers are load shed under certain conditions and the auto reclose time delay for these is 30 seconds. Second part is correct, the purpose of the load shed system is the shedding non-essential loads
reducing the load on the Main Feeder Bus to within the capacity of the standby transformers (CT4 and CT5).

Answer C Discussion

INCORRECT. First part is correct; the time delay for 1X7, 2X4 & 3X4 after a load shed is 60 seconds to reenergize. Second part is plausible because this is one of the purposes of the Main Feeder Bus Monitor Panels. MFBMP's assure the integrity of the RCP seals by insuring that seal injection and component cooling flows are regained following a loss of power

Answer D Discussion

CORRECT. The time delay for 1X7, 2X4 & 3X4 after a load shed is 60 seconds to reenergize. The purpose of the load shed system is the shedding non-essential loads reducing the load on the Main Feeder Bus to within the capacity of the standby transformers (CT4 and CT5).

Basis for meeting the KA

Requires knowledge of the relationship between a portion of the AC Distribution System (1X7, 2X4, 3X4) and Offsite Power source CT-5.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT43 Q15

Development References

ILT43 Q15
EL-PSL Obj. 03

Student References Provided

SYS062 K1.04 - AC Electrical Distribution System

Knowledge of the physical connections and/or cause-effect relationships between the ac distribution sys- tem and the following systems : (CFR: 41.2 to 41.9)

Off-site power sources ....

401-9 Comments:

Remarks/Status

Nobody missed it on 43

SYS063 2.1.23 - DC Electrical Distribution System
SYS063 GENERIC
Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10 / 43.5 / 45.2 / 45.6)

Given the following station conditions:
Initial conditions:

- All three units Reactor power = 100%

Current conditions:

- All Unit's 4160v Main Feeder Busses are de-energized
- Unit 1, 2, and 3 EOP Blackout tabs in progress

Which ONE of the following describes the required status of Unit 1 Essential Inverters per EOP Enclosure 5.38 (Restoration of Power) and why?

Unit 1's Essential Inverters _______.

A. remain energized to provide power to ES channels

B. remain energized to provide control power to 4160v

C. are de-energized to prevent inverter damage

D. are de-energized to extend the life of available batteries

---

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General Discussion

Answer A Discussion
Incorrect.: Plausible if ES Channels (are vital loads from KVIA,B,C,D) are confused with essential loads (from KI, KU, KX); vital loads must be differentiated from essential loads.

Answer B Discussion
Incorrect: Plausible if control power (ex. for breakers, switches, etc) are incorrectly assumed to be essential inverter loads

Answer C Discussion
Incorrect: Incorrect but plausible in that inverters could be damaged due to high current as input voltages start to decrease.

Answer D Discussion
Correct: Essential Inverters KI, KU, & KX DC input breakers are opened to extend battery life per direction given from the EOP SBO tab and Encl. 5.38.

Basis for meeting the KA
Requires knowledge of required actions within procedures and the correlation of the impact of high battery load on available battery capacity as the bases for actions directed in the EOP

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT45 Q49

Development References
ILT45 Q49
EAP-BO Obj. R8
EOP Encl. 5.38

Student References Provided

SYS063 2.1.23 - DC Electrical Distribution System
SYS063 GENERIC
Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10 / 43.5 / 45.2 / 45.6)

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 48 48 **B**

SYS063 K1.03 - DC Electrical Distribution System

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

Battery charger and battery ....

Given the following plant conditions:

- 3CA Battery Charger fails - output voltage = 0 VDC
- 3CA Battery voltage = 120 VDC
- 3DCB Bus voltage = 123 VDC
- Unit 1 DCA/DCB Bus voltage = 125 VDC
- Unit 2 DCA/DCB Bus voltage = 127 VDC

Which ONE of the following will automatically supply power to 3DIA panelboard?

A. 3CA Battery

B. Unit 1 DC Bus

C. 3DCB Bus

D. Unit 2 DC Bus

General Discussion

Answer A Discussion

Incorrect. Plausible because the 3CA battery will supply power to the bus if its voltage is higher than the backup source. In this case it is not. Unit 1's voltage is higher.

Answer B Discussion

Correct, The voltage from Unit 1 is higher than the 3CA battery voltage since Unit 1 is being supplied from the charger, so Unit 1 will supply power.

Answer C Discussion

Incorrect. For the Vital DC system, the 3DCB bus is not aligned to the 3DCA bus. Plausible because 3DCB Bus is aligned to backup the essential inverters

Answer D Discussion

Incorrect. Unit 2's DC Bus is not connected to Unit 3. Plausible because Unit 2 is adjacent to Unit 3 and supplies backup power to Unit 3 for other equipment such as the Unit 3 Control Rod Drive System.

Basis for meeting the KA

Requires knowledge of relationships between the control battery, battery charger, and DC Electrical System.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT44 Q75

Development References

ILT44 Q75
EL-DCD Obj. 06

Student References Provided

SYS063 K1.03 - DC Electrical Distribution System

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

Battery charger and battery ....

401-9 Comments:

Remarks/Status

SYS064 K2.03 - Emergency Diesel Generator (ED/G) System
Knowledge of bus power supplies to the following: (CFR: 41.7)
Control power ....

Given the following plant conditions:
Time = 1200

- All three units at 50% power
- ACB-3 is closed

Time = 1245

- LOCA occurs on Unit 1
- ES channels 1 and 2 actuate on Unit 1
- CT-1 lockout occurs

At Time = 1250, KHU 1 control power is supplied by ________.

Which ONE of the following completes the statement above?

A. CX Transformer from Keowee Unit 1

B. 1X Transformer from Keowee Unit 1

C. CX Transformer from Keowee Unit 2

D. 1X Transformer from 230 KV Switchyard

General Discussion

Answer A Discussion

Correct. 3 sources of power are normally available to the 1X Load Center, which in turn supplies control power to the Keowee unit. The power supplies are: 1) 1X Transformer from the 230KV switchyard; 2) 1X Transformer from KHU 1 through ACB-1; 3) CX Transformer from 1TC Switchgear. For the Underground unit, CX Transformer is the normal supply and 1X Transformer is the alternate supply. If the normal supply is lost for 36 seconds and the alternate supply is available, the alternate supply will close in. If the normal source returns during the 36 seconds, the logic returns to the normal source. In this question, KHU 1 is the underground unit. With the reactor trip and CT-1 lockout, power is lost to Unit 1 and thus 1TC (CX Transformer), but returns from KHU 1 in less than 36 seconds.

Answer B Discussion

Incorrect. . Plausible because it would be true if this was the overhead unit and a Switchyard Isolation occurred (KHU 1 running with ACB-1 closed).

Answer C Discussion

Incorrect. Plausible because it would be correct if KHU 2 were supplying power to Unit 1.

Answer D Discussion

Incorrect. Plausible because it would be correct if power were lost to 1TC (CX Transformer) for > 36 seconds.

Basis for meeting the KA

The question requires knowledge of the bus power supply to control power for the Keowee Hydro Units, which is Oconee's equivalent of Emergency Diesel Generator (ED/G) System, during various plant conditions.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References
EL-KHU Obj. 01

Student References Provided

SYS064 K2.03 - Emergency Diesel Generator (ED/G) System

Knowledge of bus power supplies to the following: (CFR: 41.7)

Control power ....

401-9 Comments:

Remarks/Status
Preview

SYS064 K3.03 - Emergency Diesel Generator (ED/G) System
Knowledge of the effect that a loss or malfunction of the ED/G system will have on the following: (CFR: 41.7 / 45.6)
ED/G (manual loads) ....

Given the following Unit 1 conditions:
Time = 1200

- Reactor power = 100%
- ACB-4 Closed
- Large Break LOCA occurs coincident with a total loss of offsite power

Time = 1205

- Keowee Hydro Unit (KHU)-2 Emergency Lockout occurs

At Time = 1210 the __(1)__ power path is being used to supply Unit 1 ECCS systems and __(2)__ LPI pumps are operating.

Which ONE of the following completes the statements above?

ASSUME NO OPERATOR ACTIONS

A. 1. Overhead
2. ONLY 2

B. 1. Overhead
2. ALL 3

C. 1. Underground
2. ONLY 2

D. 1. Underground
2. ALL 3

ILT48 ONS SRO NRC Examination QUESTION 50

50

A

General Discussion

Answer A Discussion

Correct. ACB-4 closed indicates that KHU-2 is aligned to the underground power path. Following the LOCA/LOOP, MFB would be energized by KHU-2 through the underground power path. With a subsequent loss of KHU-2, retransfer to startup logic would transfer power to the overhead power path which would be supplied by KHU-1.
Although there are 3 LPI pumps, only the A and B are ECCS pumps therefore only 2 would be operating.

Answer B Discussion

Incorrect. First part is correct. Second part is plausible since there are 3 LPI pumps and other ECCS systems with 3 major components require all 3 (HPI, LPSW, RBCU's).

Answer C Discussion

Incorrect. ACB-4 closed indicates that KHU-2 is aligned to the underground power path. Following the LOCA/LOOP, MFB would be energized by KHU-2 through the underground power path. With a subsequent loss of KHU-2, retransfer to startup logic would transfer power to the overhead power path which would be supplied by KHU-1. However, if the Startup source were not available KHU-1 would be aligned to the Underground Powerpath by closing ACB-3.
Second part is correct

Answer D Discussion

Incorrect. ACB-4 closed indicates that KHU-2 is aligned to the underground power path. Following the LOCA/LOOP, MFB would be energized by KHU-2 through the underground power path. With a subsequent loss of KHU-2, retransfer to startup logic would transfer power to the overhead power path which would be supplied by KHU-1. However, if the Startup source were not available KHU-1 would be aligned to the Underground Powerpath by closing ACB-3.
Second part is plausible since there are 3 LPI pumps and other ECCS systems with 3 major components require all 3 (HPI, LPSW, RBCU's).

Basis for meeting the KA

Our "EDG system" is our Keowee Hydro Units. This question requires knowledge of the effect that a loss of one of the KHU's will have on ONS plant loads and the other KHU. You have to know that when KHU is supplying emergency power to CT-4 following a loca/loop and that KHU is lost, the retransfer to startup will align the other KHU through the overhead power path to the units Main Feeder Buses which are supplying ECCS loads.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	ILT46 Q49

Development References

ILT46 Q49
EL-PSL Obj 07
PNS-LPI

Student References Provided

SYS064 K3.03 - Emergency Diesel Generator (ED/G) System
Knowledge of the effect that a loss or malfunction of the ED/G system will have on the following: (CFR: 41.7 / 45.6)
ED/G (manual loads) ....

401-9 Comments:

Remarks/Status

SYS073 K4.01 - Process Radiation Monitoring (PRM) System
Knowledge of PRM system design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)
Release termination when radiation exceeds setpoint ....

Given the following Unit 1 conditions:

- Unit 1 startup in progress
- RB Purge in progress
- 1RIA-45 (Norm Vent Gas) determined to be inoperable

1) Automatic termination of RB Purge operation due to increasing activity __(1)__ available.

2) If automatic termination of RB Purge occurs, __(2)__ will close.

Which ONE of the following completes the statements above?

A. 1. is
2. 1PR-2 through 1PR-5 ONLY

B. 1. is
2. 1PR-1 through 1PR-6

C. 1. is NOT
2. 1PR-2 through 1PR-5 ONLY

D. 1. is NOT
2. 1PR-1 through 1PR-6

ILT48 ONS SRO NRC Examination QUESTION 51

51

A

General Discussion

Answer A Discussion

Correct In case of a failure of RIA-45 HIGH alarm, RIA-46 HIGH alarm (via the switchover function) will actuate the required interlock functions which will trip the RB Purge Fan and close 1PR-2 through 1PR-5.

Answer B Discussion

Incorrect, First part is correct. In case of a failure of RIA-45 HIGH alarm, RIA-46 HIGH alarm (via the switchover function) will actuate the required interlock functions.
Second part is incorrect. Plausible since 1PR-1 and 1PR-6 are containment isolation valves and are closed by an ES actuation.

Answer C Discussion

Incorrect. First part is plausible since 1RIA-45 provides the normal means of automatic isolation of RB purge based on increasing activity therefore it would be plausible to assume that if 1RIA-45 did not auto terminate RB Purge then manual termination would be required.
Second part is correct.

Answer D Discussion

Incorrect. First part is plausible since 1RIA-45 provides the normal means of automatic isolation of RB purge based on increasing activity therefore it would be plausible to assume that if 1RIA-45 did not auto terminate RB Purge then manual termination would be required.
Second part is incorrect. Plausible since 1PR-1 and 1PR-6 are containment isolation valves and are closed by an ES actuation.

Basis for meeting the KA

Requires knowledge of the effect that a loss of RIA-45 will have on the availability of automatic termination if RIA setpoint is exceeded.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	ILT44 Q41

Development References

ILT44 Q41
RAD-RIA obj 08
ARG 1SA-8/B9

Student References Provided

SYS073 K4.01 - Process Radiation Monitoring (PRM) System
Knowledge of PRM system design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)
Release termination when radiation exceeds setpoint ....

401-9 Comments:

Remarks/Status

SYS073 K5.03 - Process Radiation Monitoring (PRM) System
Knowledge of the operational implications as they apply to concepts as they apply to the PRM system: (CFR: 41.5 / 45.7)
Relationship between radiation intensity and exposure limits ....

Given the following plant conditions:
Time = 1200

- Spent Fuel Pool level = 0.1 foot stable

Time = 1215

- Spent Fuel Pool level = -3.4 feet decreasing
- 1RIA-6 (Spent Fuel Pool Area Monitor) in HIGH alarm
- 1RIA-41 (Spent Fuel Pool Building Gas) in HIGH alarm
- An AO is being dispatched to the SFP area to investigate the cause.
- The AO's dose for this year is 525 mrem
- The AO has NOT received a dose extension for this year

Which ONE of the following is the MAXIMUM TEDE dose (mrem) allowed for the AO while performing the assigned task?

A. 1475

B. 4475

C. 5,000

D. 10,000

General Discussion

Answer A Discussion

Incorrect. Plausible since this would be correct if EDL's were not in effect.

Answer B Discussion

Incorrect. Plausible since this would be correct under the misconception that the 5 rem EDL limit included normal occupational exposure for the associated year.

Answer C Discussion

Correct. 5 rem is the EDL limit for individual exposure

Answer D Discussion

Incorrect. Plausible since this is the limit for all activities during an EDL event and could be correct if the Shift Manager were consulted to allow exceeding the 5 Rem or if the actions were specifically to protect valuable property.

Basis for meeting the KA

Meets the KA because a process monitor is being used to determine the operational implications of exposure limits. RIA-41 is a process monitor which is indicating an increase in radiation intensity and that increase in intensity is (in part) a determining factor for what exposure limits apply. Since there is no direct relationship between radiation exposure limits and radiation intensity (e.g. dose limits are not variable based on radiation intensity), making the connection between an accident that causes increased radiation intensity which also impacts exposure limits is used to match the KA.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References

EAP-TCA obj R6
AP/35

Student References Provided

SYS073 K5.03 - Process Radiation Monitoring (PRM) System

Knowledge of the operational implications as they apply to concepts as they apply to the PRM system: (CFR: 41.5 / 45.7)

Relationship between radiation intensity and exposure limits ....

401-9 Comments:

Remarks/Status

SYS076 A1.02 - Service Water System (SWS)

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

Reactor and turbine building closed cooling water temperatures. ....

Given the following Unit 3 conditions:

Time = 1200

- Reactor power = 100%
- 3A Letdown cooler tube leak rate = 0.1 gpm

Time = 1230

- 3A Letdown cooler tube leak rate = 5.0 gpm

1) LPSW flow to the Component Coolers would be required to__(1)__ in order to maintain 3A Letdown Cooler Component Cooling outlet temperature constant from 1200 – 1230.

2) The MINIMUM RCS letdown temperature that will result in an AUTOMATIC isolation of letdown is __(2)__°F.

Which ONE of the following completes the statements above?

A. 1. increase
2. 130

B. 1. increase
2. 135

C. 1. remain approximately the same
2. 130

D. 1. remain approximately the same
2. 135

General Discussion

Answer A Discussion

First part is correct. Second part is plausible since this is the high letdown temperature alarm setpoint.

Answer B Discussion

Correct. Since RCS is at a higher temperature and pressure than Component Cooling and the Letdown Coolers are located before the RCS pressure breakdown orifice in the letdown line the tube leakage would be RCS leaking into the CC system (CC pump disch pressure approx. 150 psig), That means that CC temperature would be increasing requiring LPSW flow to increase in order to maintain CC temp constant. If letdown temperature reaches 135 degrees, 1HP-5 will automatically close isolating letdown.

Answer C Discussion

First part is plausible for two reasons. If the candidate confuses the other closed loop cooling system (RCW) as the Letdown cooler cooling medium then they would pick LPSW flow to remain unchanged. However the more likely plausibility comes from the location of the Letdown coolers in the letdown line. If the coolers were downstream of the letdown orifice then leakage would be by way of CC leaking into the RCS which would make LPSW flow remaining approximately unchanged the correct choice.
Second part is plausible since this is the high letdown temperature alarm setpoint.

Answer D Discussion

First part is plausible for two reasons. If the candidate confuses the other closed loop cooling system (RCW) as the Letdown cooler cooling medium then they would pick LPSW flow to remain unchanged. However the more likely plausibility comes from the location of the Letdown coolers in the letdown line. If the coolers were downstream of the letdown orifice then leakage would be by way of CC leaking into the RCS which would make LPSW flow remaining approximately unchanged the correct choice.
Second part is correct.

Basis for meeting the KA

The SWS is our LPSW system. This question requires the ability to predict a parameters response (3A Letdown Cooler CC outlet temp) associated with operating the LPSW controls since LPSW is the cooling medium for the CC system. CC (Component Cooling) is the Reactor closed loop cooling water system at Oconee,

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References
PNS-CC Obj 03, 04, 06 PNS-HPI

Student References Provided

SYS076 A1.02 - Service Water System (SWS)

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

Reactor and turbine building closed cooling water temperatures. ....

401-9 Comments:

Remarks/Status

SYS078 A3.01 - Instrument Air System (IAS)
Ability to monitor automatic operation of the IAS, including: (CFR: 41.7 / 45.5)
Air pressure ....

Given the following plant conditions:
Initial conditions:

- Time = 0400
- Primary IA compressor tripped

Current conditions:

- Time = 0405
- Instrument Air pressure = 91 psig decreasing

At Time = 0405…

1) Auxiliary IA Compressors are __(1)__.

2) Backup IA compressors are __(2)__.

Which ONE of the following completes the statements above?

A. 1. OFF
2. OFF

B. 1. Operating
2. OFF

C. 1. OFF
2. Operating

D. 1. Operating
2. Operating

General Discussion

Backup IA Compressors switches are maintained in STBY 1 during normal operation.
Backup IA Compressors in STBY 1 start at ≈ 93 psig.
Unit 1 Aux IA Compressor starts at ≈ 88 psig.
Sullair Service Air Compressors start at 95 psig.
SA-141 (SA to IA Controller) automatically regulates IA header pressure to 85 psig.
Diesel Air Compressors in Auto start at ≈ 90 psig.

Answer A Discussion

Incorrect:. First part is correct. Second part is incorrect. The Backup IA Compressors are normally in STBY 1 and auto start at 93 psig decreasing. Plausible because it would be correct if the Backup IA Compressor switches were in STBY 2. In STBY 2 the auto start setpoint is 90 psig decreasing.

Answer B Discussion

Incorrect:. First part is incorrect. Plausible because it would be correct for the Sullair Service Air Compressors, which auto start at 95 psig decreasing SA pressure. Second part is incorrect. The Backup IA Compressors are normally in STBY 1 and auto start at 93 psig decreasing. Plausible because it would be correct if the Backup IA Compressor switches were in STBY 2. In STBY 2 the auto start setpoint is 90 psig decreasing.

Answer C Discussion

Correct:. Backup IA compressors started at 93 psig and the AIA compressors are still off because they start at 88 psig.

Answer D Discussion

Incorrect.First part is incorrect. Plausible because it would be correct for the Sullair Service Air Compressors, which auto start at 95 psig decreasing SA pressure. Second part is correct.

Basis for meeting the KA

Question requires knowledge of how Instrument Air System components respond to changes in instrument air pressure.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT41 Q52

Development References

ILT41 Q52
SSS-IA Obj. 16 & 25
AP/22

Student References Provided

SYS078 A3.01 - Instrument Air System (IAS)
Ability to monitor automatic operation of the IAS, including: (CFR: 41.7 / 45.5)
Air pressure ....

401-9 Comments:

Remarks/Status

SYS103 A2 04 - Containment System

Ability to (a) predict the impacts of the following malfunctions or operations on the containment 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)

Containment evacuation (including recognition of the alarm) ....

Given the following Unit 1 conditions:
Time = 1200

- Reactor in MODE 5
- RCS Loops dropped
- Pressurizer level = 340” stable
- RB Purge in progress
- Reactor Building Normal Sump is being pumped

Time = 1205

- 1RIA-48 (Reactor Building Iodine) in HIGH alarm
- 1RIA-49 (Reactor Building Gas) in HIGH alarm

1) The Containment Evacuation alarm __(1)__ AUTOMATICALLY actuate.

2) __(2)__ will provide guidance to complete the system isolation initiated by 1RIA-49.

Which ONE of the following completes the statements above?

A. 1. will
2. OP/1/A/1102/014 (RB Purge)

B. 1. will
2. OP/1/A/1104/007 (LWD System)

C. 1. will NOT
2. OP/1/A/1102/014 (RB Purge)

D. 1. will NOT
2. OP/1/A/1104/007 (LWD System)

General Discussion

Answer A Discussion

Incorrect. First part is correct. Second part is incorrect. Plausible since it would be correct on a 1RIA-45 high alarm.

Answer B Discussion

CORRECT. 1RIA-49 HIGH alarm does actuate the RB Evacuation alarm. 1RIA-49 HIGH alarm also closes 1LWD-2 to isolate the RBNS. The procedure used to pump the RBNS is OP/1/A/1104/007 (LWD System) and provides guidance to ensure the RBNS Pumps are off and 1LWD-1 is closed.

Answer C Discussion

Incorrect. First part is incorrect. Plausible since some other RIA's in the RB do NOT actuate the Containment Evacuation alarm (Ex. RIA-3, 57, 58). Second part is incorrect. Plausible since it would be correct on a 1RIA-45 high alarm.

Answer D Discussion

Incorrect. First part is incorrect. Plausible since some other RIA's in the RB do NOT actuate the Containment Evacuation alarm (Ex. RIA-3, 57, 58). Second part is correct.

Basis for meeting the KA

Question requires predicting actuation of the Containment Evacuation alarm and using procedures to complete the isolation of the RBNS.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	ILT46 Q54

Development References

ILT46 Q54
RAD-RIA Obj. 08
OP/1/A/1104/007

Student References Provided

SYS103 A2 04 - Containment System

Ability to (a) predict the impacts of the following malfunctions or operations on the containment 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)

Containment evacuation (including recognition of the alarm) ....

401-9 Comments:

Remarks/Status

PREVIEW

SYS001 K2.03 - Control Rod Drive System
Knowledge of bus power supplies to the following: (CFR: 41.7)
One-line diagram of power supplies to logic circuits ....

Which ONE of the following describes ALL of the MCCs which supply power to Unit 1's Control Rod Drive System?

A. 1X9 and 2X1

B. 1X9 and 2X2

C. 1X1 and 2X9

D. 1X1 and 3X9

ILT48 ONS SRO NRC Examination QUESTION 56

56

A

General Discussion

Answer A Discussion

Correct. Normal CRD supply is 1X9. Alternate supply is 2X1.

Answer B Discussion

Incorrect. Plausible because 1X9 is correct and 2X2 is plausible because 2X2 is Unit 3s Alternate supply.

Answer C Discussion

Incorrect. Plausible because would be correct for Unit 2.

Answer D Discussion

Incorrect. Plausible if the candidate had the misconception that Unit 3 backed up Unit 1. That misconception is plausible since the Unit 3 DC system does back up Unit 1s DC system.

Basis for meeting the KA

Question requires knowledge of the normal and alternate power supplies for the CRD system which includes the logic circuits used to ensure power to CRD's is available or removed as required.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT41 Q56

Development References

ILT41 Q56
IC-RPS obj R27

Student References Provided

SYS001 K2.03 - Control Rod Drive System
Knowledge of bus power supplies to the following: (CFR: 41.7)
One-line diagram of power supplies to logic circuits ....

401-9 Comments:

Remarks/Status

SYS002 K6.03 - Reactor Coolant System (RCS)
Knowledge of the effect or a loss or malfunction on the following RCS components: (CFR: 41.7 / 45.7)
Reactor vessel level indication ....

Given the following Unit 1 conditions:

- Reactor in MODE 5
- Reactor Vessel level = 80" stable
- Reactor Vessel Cold Leg Ultrasonic level NOT available

1) In accordance with SD 1.3.5 (Shutdown Protection Plan), reducing the reactor vessel level to 40" __(1)__ allowed.

2) Improper venting of the RCS results in RCS pressure greater than Containment pressure and LT-5 readings that are __(2)__ than actual.

Which ONE of the following completes the statements above?

A. 1. is
   2. greater

B. 1. is
   2. less

C. 1. is NOT
   2. greater

D. 1. is NOT
   2. less

ILT48 ONS SRO NRC Examination QUESTION 57

57

C

General Discussion

Answer A Discussion

Incorrect. First part is incorrect. Plausible since it would be correct if level were not going below 50 inches.
Second part is correct. LT-5 contains a dp transmitter filled reference leg that is open to the containment atmosphere. If RCS pressure is greater than containment pressure, LT-5 will indicate greater than actual level.

Answer B Discussion

Incorrect. First part is incorrect. Plausible since it would be correct if level were not going below 50 inches.
Second part is incorrect. LT-5 contains a dp transmitter filled reference leg that is open to the containment atmosphere. If RCS pressure is greater than containment pressure, LT-5 will indicate greater than actual level. It would be correct if containment pressure were greater than RCS pressure.

Answer C Discussion

Correct.
First part is correct. SD 1.3.5 requires that all 4 RV level indications be operable prior to going below 50" RV level.
Second part is correct. LT-5 contains a dp transmitter filled reference leg that is open to the containment atmosphere. If RCS pressure is greater than containment pressure, LT-5 will indicate greater than actual level.

Answer D Discussion

Incorrect. First part is correct. SD 1.3.5 requires that all 4 RV level indications be operable prior to going below 50" RV level.
Second part is incorrect. LT-5 contains a dp transmitter filled reference leg that is open to the containment atmosphere. If RCS pressure is greater than containment pressure, LT-5 will indicate greater than actual level. It would be correct if containment pressure were greater than RCS pressure.

Basis for meeting the KA

Requires knowledge of RV level restrictions as a result of a loss of one of the RV level indications. With the loss of the Reactor Vessel cold leg Ultrasonic level, the RCS drain procedure will not allow reducing the reactor vessel level < 50".

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References

ADM-SPP Obj 10
SD 1.3.5
CP-RCD Obj 10

Student References Provided

SYS002 K6.03 - Reactor Coolant System (RCS)
Knowledge of the effect or a loss or malfunction on the following RCS components: (CFR: 41.7 / 45.7)
Reactor vessel level indication ....

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 58 58 **D**

SYS011 K3.03 - Pressurizer Level Control System (PZR LCS)
Knowledge of the effect that a loss or malfunction of the PZR LCS will have on the following: (CFR: 41.7 / 45.6)
PZR PCS ....

Given the following Unit 1 conditions:
Time = 1200

- Plant is stable following a Reactor trip
- Pressurizer (Pzr) level = 100” stable

Time = 1230

- 1HP-120 fails closed
- Pressurizer level decreasing at 1” per minute

1) Assuming no operator actions, the EARLIEST time that Pzr heaters will become unavailable is __(1)__.

2) Based on the conditions above, AP/14 (Loss of Normal HPI Makeup And/Or RCP Seal Injection) __(2)__ direct closing 1HP-5 while 1HP-120 is being repaired.

A. 1. 1245
 2. will

B. 1. 1245
 2. will NOT

C. 1. 1250
 2. will

D. 1. 1250
 2. will NOT

ILT48 ONS SRO NRC Examination QUESTION 58

58

D

General Discussion

Answer A Discussion

Incorrect. First part is incorrect. Plausible since this time equates to 85" which would be correct if asking about SSF Pzr heaters.
Second part is incorrect. Plausible since AP/14 does direct closing 1HP-5 when suction is lost to the HPI Pumps.

Answer B Discussion

Incorrect. First part is incorrect. Plausible since this time equates to 85" which would be correct if asking about SSF Pzr heaters.
Second part is correct

Answer C Discussion

Incorrect. First part is correct.
Second part is incorrect. Plausible since AP/14 does direct closing 1HP-5 when suction is lost to the HPI Pumps.

Answer D Discussion

Correct. This time equates to 80" Pzr level which is the low level cutoff for the Pzr heaters. AP/14 will not direct closing 1HP-5 under the given conditions.

Basis for meeting the KA

Required knowledge of the effect that 1HP-120 failing closed will have on Pressurizer heaters.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References

EAP-APG R9
PNS-Pzr obj R7
AP/14

Student References Provided

SYS011 K3.03 - Pressurizer Level Control System (PZR LCS)
Knowledge of the effect that a loss or malfunction of the PZR LCS will have on the following: (CFR: 41.7 / 45.6)
PZR PCS ....

401-9 Comments:

Remarks/Status

SYS016 A3.02 - Non-Nuclear Instrumentation System (NNIS)
Ability to monitor automatic operation of the NNIS, including: (CFR: 41.7 / 45.5)
Relationship between meter readings and actual parameter value ....

Given the following Unit 1 conditions:

- Main Steam Line Break on the 1A Steam Generator
- Reactor Building pressure = 6.3 psig very slowly decreasing
- Indicated 1B SG XSUR level = 75” decreasing
- ALL Subcooling Margins = 65°F stable

1) Actual 1B SG level is __(1)__ indicated XSUR level.

2) 1B SG level __(2)__ AUTOMATICALLY be controlled at the desired level in accordance with Rule 7 (SG Feed Control).

Which ONE of the following completes the statements above?

A. 1. lower than
2. will

B. 1. lower than
2. will NOT

C. 1. approximately the same as
2. will

D. 1. approximately the same as
2. will NOT

General Discussion

Answer A Discussion

Incorrect: First part is correct.
Second part is plausible for three reasons. 1. If SG pressure did not exceed 3 psig it would be correct. 2. It would be plausible to believe that the SG level control system setpoint would change when ACC conditions exist and therefore control at the level designated in Rule 7. 3. SG level will be automatcially controlled in this case, only it will be controlled at 30" instead of 60" as required by Rule 7.

Answer B Discussion

Correct. Since XSUR is not temperature compensated, as RB temperatures increase, actual level becomes lower than indicated level. As RB pressure exceeds 3 psig, ACC conditions are determined to exist and Rule 7 will direct controlling available steam generators at 60" XSUR. This requires manually controlling level since auto level control will attempt to control at 30" XSUR.

Answer C Discussion

Incorrect: First part is plausible since it would be correct if asking about SG Operating Range level since it is temperature compensated.
Second part is plausible for three reasons. 1. If SG pressure did not exceed 3 psig it would be correct. 2. It would be plausible to believe that the SG level control system setpoint would change when ACC conditions exist and therefore control at the level designated in Rule 7. 3. SG level will be automatcially controlled in this case, only it will be controlled at 30" instead of 60" as required by Rule 7

Answer D Discussion

Incorrect: First part is plausible since it would be correct if asking about SG Operating Range level since it is temperature compensated.
Second part is correct.

Basis for meeting the KA

Requires the ability to determine the relationship between actual SG level and indicated SG level following a MSLB inside containment. Also requires the ability to monitor automatic operation of SG level control system and determine if it is controlling at the desired level following a MSLB.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References	Student References Provided
EAP-EHT obj R13 Rule 7	

SYS016 A3.02 - Non-Nuclear Instrumentation System (NNIS)
Ability to monitor automatic operation of the NNIS, including: (CFR: 41.7 / 45.5)
Relationship between meter readings and actual parameter value ....

401-9 Comments:	Remarks/Status

SYS035 K4.01 - Steam Generator System (S/GS)
Knowledge of S/GS design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)
S/G level control ....

Given the following Unit 1 conditions:
Initial conditions:

- Reactor power = 100%

Current conditions:

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

Which ONE of the following is the Steam Generator level that will be automatically maintained?

A. 25 inches Startup Range

B. 30 inches XSUR

C. 50% Operating Range

D. 240 inches XSUR

General Discussion

Answer A Discussion

Incorrect. Plausible since it would be correct if Main Feedwater were still available

Answer B Discussion

Correct. With Vacuum at 18.5 inches, Main Feedwater pumps have already tripped on low vacuum. Since there are still 2 RCP's operating EFDW will control level at 30" on XSUR.

Answer C Discussion

Incorrect. Plausible since this would be correct if Main Feedwater were still available and all 4 RCP's were tripped.

Answer D Discussion

Incorrect. Plausible since it would be correct if both 1TA and 1TB had been de-energized.

Basis for meeting the KA

Requires knowledge of automatic Steam Generator level control design.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	ILT46 Q45

Development References

ILT46 Q45
CF-EF Obj 31
CF-FW Obj 11

Student References Provided

SYS035 K4.01 - Steam Generator System (S/GS)

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

S/G level control ....

401-9 Comments:

Remarks/Status

New K/A

SYS045 A1.05 - Main Turbine Generator (MT/G) System

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

Expected response of primary plant parameters (temperature and pressure) following T/G trip ....

Given the following Unit 1 conditions:

- Power escalation in progress
- Reactor power = 25% slowly increasing
- Main Condenser vacuum 27” Hg and degrading

1) The Main Turbine low vacuum trip setpoint is __(1)__ inches Hg.

2) After restoring vacuum to normal and stabilizing from the transient, Tave will be approximately __(2)__ degrees F.

Which ONE of the following completes the statements above?

A. 1. 19
   2. 555

B. 1. 19
   2. 579

C. 1. 21.75
   2. 555

D. 1. 21.75
   2. 579

General Discussion

Answer A Discussion

Incorrect. First part is plausible since that is the low vacuum trip setpoint for the Main Feedwater pumps.
Second part is plausible since it would be correct for Turbine Trips that occur with Rx Power >29.75% power.

Answer B Discussion

Incorrect. First part is plausible since that is the low vacuum trip setpoint for the Main Feedwater pumps.
Second part is correct.

Answer C Discussion

Incorrect. First part is correct.
Second part is plausible since it would be correct for Turbine Trips that occur with Rx Power >29.75% power.

Answer D Discussion

Correct. The low vacuum trip setpoint for the Main Turbine is 21.75" Hg. Since the Turb/Rx trip is not yet armed the Rx will not trip as a result of the Main Turbine trip therefore the resulting Tave would remain approximately 579 degrees once the plant stabilized.

Basis for meeting the KA

Requires the ability to predict changes in Tave following a Main Turbine trip a low in the power range.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References

STG-MT Obj 16
CF-FWPT
IC-RPS

Student References Provided

SYS045 A1.05 - Main Turbine Generator (MT/G) System

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

Expected response of primary plant parameters (temperature and pressure) following T/G trip ....

401-9 Comments:

Remarks/Status

SYS055 K1.06 - Condenser Air Removal System (CARS)

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

PRM system ....

Given the following Unit 1 conditions:

- Reactor power = 35% stable
- Air Ejector off-gas activity increasing
- Local RP survey indicates significant increase in 1B Main Steam Line activity

1) 1RIA-40 (Air Ejector Off Gas) __(1)__ be used to determine Primary to Secondary leak rate.

2) In accordance with the SGTR tab, the MAXIMUM RCS temperature allowed when isolating the 1B SG is __(2)__ degrees F.

Which ONE of the following completes the statements above?

A. 1. can
2. 525

B. 1. can
2. 532

C. 1. can NOT
2. 525

D. 1. can NOT
2. 532

General Discussion

Answer A Discussion

Incorrect.
First part is correct. 1RIA-40 can be used to determine Primary to Secondary leak rate. Second part is incorrect. Plausible since the band to isolate the SG is 525 to 532 degrees as directed by the SGTR tab.

Answer B Discussion

Correct.
First part is correct. 1RIA-40 can be used to determine Primary to Secondary leak rate. Second part is correct. The SGTR tab directs cooling the RCS to a band of 525 to 532 degrees prior to isolating the SG.

Answer C Discussion

Incorrect. First part is incorrect. Plausible since it would be correct if asking about RIA-59/60. Per the SGTR tab, RIA-59/60 cannot be used to determine leak rate with Reactor power is < 40%.
Second part is incorrect. Plausible since the band to isolate the SG is 525 to 532 degrees as directed by the SGTR tab.

Answer D Discussion

Incorrect. First part is incorrect. Plausible since it would be correct if asking about RIA-59/60. Per the SGTR tab, RIA-59/60 cannot be used to determine leak rate with Reactor power is < 40%. Second part is correct.

Basis for meeting the KA

This question requires knowledge of the cause-effect relationship between increasing activity in the air ejector off gas system and Process Radiation Monitor system use by the operators.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	MODIFIED	2009B Q9

Development References

EAP-AP31 Obj R1, R9
EAP-SGTR Obj 28,29,30
2009B Q9

Student References Provided

SYS055 K1.06 - Condenser Air Removal System (CARS)

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

PRM system ....

401-9 Comments:

Remarks/Status

SYS071 A4.10 - Waste Gas Disposal System (WGDS)

Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)

WGDS sampling ....

In accordance with OP/1-2/A/1104/018 (GWD System)…

1) A newly isolated GWD tank must be sampled for Hydrogen within __(1)__ hours.

2) If Hydrogen concentration in a GWD tank is 3.5%, the Hydrogen concentration must be reduced to $\leq$ 3% within __(2)__ hours.

Which ONE of the following completes the statements above?

A. 1. 24
   2. 24

B. 1. 24
   2. 48

C. 1. 48
   2. 24

D. 1. 48
   2. 48

General Discussion

Answer A Discussion

Incorrect. First part is correct. Second part is incorrect but plausible since it would be correct if hydrogen concentration were greater than 4%.

Answer B Discussion

Correct. IAW limits and precautions of 1104/18, an isolated GWD tank must be sampled for Hydrogen within 24 hours of being isolated. If H2 concentration in a tank exceeds 3% but is less than or equal to 4% volume, the concentration must be reduced to less than or equal to 3% within 48 hours.

Answer C Discussion

Incorrect. First part is plausible since 48 hours is the time allowed to reduce Hydrogen pressure to less than or equal to 3% when hydrogen is > 3% but less than or equal to 4%. Second part is incorrect but plausible since it would be correct if hydrogen concentration were greater than 4%.

Answer D Discussion

Incorrect. First part is plausible since 48 hours is the time allowed to reduce Hydrogen pressure to less than or equal to 3% when hydrogen is > 3% but less than or equal to 4%. Second part is correct.

Basis for meeting the KA

Requires the ability to monitor H2 concentration and perform actions related to operation of the GWD tanks and sampling of the tanks. Actions for tank sampling, isloations ,and additions are initiated from the Control Room.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References

WE-GWD obj 11
1104/18

Student References Provided

SYS071 A4.10 - Waste Gas Disposal System (WGDS)

Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)

WGDS sampling ....

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 64 64 **D**

SYS079 A2.01 - Station Air System (SAS)

Ability to (a) predict the impacts of the following malfunctions or operations on the SAS; 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)

Cross-connection with IAS ....

Given the following Unit 1 conditions:

- Reactor power = 100%
- Instrument Air (IA) header pressure = 90 psig decreasing

1) Unit __(1)__ AP/22 (Loss of Instrument Air) will dispatch an operator to ensure Diesel Air Compressors are operating.

2) IF IA header pressure reaches 85 psig the Service Air compressors will __(2)__ the instrument air header.

A. 1. one
   2. manually be aligned to

B. 1. two
   2. manually be aligned to

C. 1. one
   2. automatically begin to supply

D. 1. two
   2. automatically begin to supply

General Discussion

Answer A Discussion

Incorrect. First part is incorrect. Plausible since there are definitely actions that need to be taken outside of the Control Room during a loss of IA on Unit 1, however it is the Unit 2 AP/22 that directs those actions. Second part is incorrect. Plausible since other system cross connects directed by AP's are performed manually (example: cross connecting LPSW systems on loss of LPSW).

Answer B Discussion

Incorrect. First part is correct. Second part is incorrect. Plausible since other system cross connects directed by AP's are performed manually (example: cross connecting LPSW systems on loss of LPSW).

Answer C Discussion

Incorrect. First part is incorrect. Plausible since there are definitely actions that need to be taken outside of the Control Room during a loss of IA on Unit 1, however it is the Unit 2 AP/22 that directs those actions.
Second part is correct.

Answer D Discussion

Correct. The Unit 2 AP/22 directs actions outside the control room on a loss of IA. The setpoint for SA-141 is 85 psig. When that pressure is reached, SA-141 begins to regulate IA pressure using the Service Air System.

Basis for meeting the KA

Requires the ability to predict the impact of decreasing IA pressure on SA system more specifically the impact of reaching the cross connect setpoint. It also requires using procedures to mitigate the consequences of decreasing IA pressure when operation of SA-141 will occur. Some of these actions are specifically associated with the fact that the IA and SAS have cross connected (ex. Ensuring DAC is operating and start if necessary).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	MODIFIED	ILT46 Q65

Development References

ILT46 Q65
EAP-APG R9
AP/22

Student References Provided

SYS079 A2.01 - Station Air System (SAS)

Ability to (a) predict the impacts of the following malfunctions or operations on the SAS; 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)

Cross-connection with IAS ....

401-9 Comments:

Remarks/Status

SYS086 K5.03 - Fire Protection System (FPS)

Knowledge of the operational implication of the following concepts as they apply to the Fire Protection System: (CFR: 41.5 / 45.7)

Effect of water spray on electrical components ....

Given the following Unit 1 conditions:

- Reactor power = 100%
- An Active fire has been reported in 1TA switchgear

1) De-energizing 1TA switchgear will result in a loss of the 1A1 and the __(1)__ Reactor Coolant Pumps.

2) In accordance with the "Fire Plan", a water fog __(2)__ be used on the switchgear to fight the fire.

Which ONE of the following completes the statements above?

A. 1. 1A2
2. can

B. 1. 1A2
2. can NOT

C. 1. 1B1
2. can

D. 1. 1B1
2. can NOT

General Discussion

Answer A Discussion

Incorrect. First part is plausible since it would be logical to assume that the two "A" loop RCP's were powered from 1TA switchgear. Second part is correct.

Answer B Discussion

Incorrect. First part is plausible since it would be logical to assume that the two "A" loop RCP's were powered from 1TA switchgear. Second part is plausible since water conducts electricity and therefore there would be a concern of electrical shock. Also since water on metal promotes oxidation it would also be plausible to believe it is prohibited in order to not cause further damage to the switchgear. And lastly it is plausible because a water stream cannot be used.

Answer C Discussion

Correct. The 1A1 and 1B1 RCP's are powered from the 1TA switchgear. A water fog is allowed to be used on electrical switchgear fires.

Answer D Discussion

Incorrect. First part is correct. Second part is plausible since water conducts electricity and therefore there would be a concern of electrical shock. Also since water on metal promotes oxidation it would also be plausible to believe it is prohibited in order to not cause further damage to the switchgear. And lastly it is plausible because a water stream cannot be used.

Basis for meeting the KA

Per CE, OK to ask about use of fog pattern with water on electrical equip. This question requires knowledge of the use of water fog on electrical equipment and the effect of de-energizing switchgear when using a water fog to extinguish a fire.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	MODIFIED	ILT42 Q99

Development References

ILT42 Q99
RCP pwr supply
Fire Plan
TA-OE Obj R2

Student References Provided

SYS086 K5.03 - Fire Protection System (FPS)

Knowledge of the operational implication of the following concepts as they apply to the Fire Protection System: (CFR: 41.5 / 45.7)

Effect of water spray on electrical components ....

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 66 66 **B**

GEN2.1 2.1.13 - GENERIC - Conduct of Operations
Conduct of Operations
Knowledge of facility requirements for controlling vital/controlled access. (CFR: 41.10 / 43.5 / 45.9 / 45.10)

---

1) The MAXIMUM number of visitors you can escort into a Vital Area is __(1)__.

2) If a Site Assembly were to occur while you were in a Vital Area with visitors you should report to your assembly location __(2)__.

Which ONE of the following completes the statements above?

A. 1. 5
2. once you have returned the visitors to Security at the PAP

B. 1. 5
2. immediately taking the visitors with you

C. 1. 10
2. once you have returned the visitors to Security at the PAP

D. 1. 10
2. immediately taking the visitors with you

ILT48 ONS SRO NRC Examination QUESTION 66

66

B

General Discussion

Answer A Discussion

First part is correct. Second part is plausible since that is where visitors are returned to once they have finished with being inside the protected area.

Answer B Discussion

Correct. You can only escort 5 visitors into Oconee Vital areas and if a site assembly were to occur while you were escorting visitors you would take the visitors to your assembly location with you.

Answer C Discussion

Incorrect. First part is plausible since it is correct for other areas inside the Protected Area. Second part is plausible since that is where visitors are returned to once they have finished with being inside the protected area.

Answer D Discussion

Incorrect. First part is plausible since it is correct for other areas inside the Protected Area. Second part is correct.

Basis for meeting the KA

CE allowed asking "initial training" information since there are not items specific to license duties related to this KA. Question requires knowledge of facility requirements for limits on personnel you can escort into a vital area.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

EOP-SEP R10,11,12
Annual Update
RP/1000/009

Student References Provided

GEN2.1 2.1.13 - GENERIC - Conduct of Operations
Conduct of Operations
Knowledge of facility requirements for controlling vital/controlled access. (CFR: 41.10 / 43.5 / 45.9 / 45.10)

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 67 67 **D**

GEN2.1 2.1.41 - GENERIC - Conduct of Operations
Conduct of Operations
Knowledge of the refueling process. (CFR: 41.2 / 41.10 / 43.6 / 45.13)

Given the following Unit 3 conditions:

- Reactor in MODE 6
- Refueling in progress

Which ONE of the following describes the source range NI requirements while refueling the reactor in accordance with OP/3/A/1502/007 (Operations Defueling/Refueling Responsibilities)?

A. Reactor Operator can use any one source range NI

B. Reactor Operator can use any two source range NI's

C. Reactor Engineering will specify the one required source range NI

D. Reactor Engineering will specify the two required source range NI's

---

General Discussion

Answer A Discussion

Incorrect: Plausible since it would be correct if fuel handling was not in progress. Only 1 SR NI is required to be operable while shutdown when fuel handling is not in progress.

Answer B Discussion

Incorrect: Plausible since the limits and precautions section of 1502/007 (Operations Defueling/Refueling Responsibilities) states "Any combination of two Source Range NI's may be used for defueling." However the two must be selected by Reactor Engineering per the body of the procedure (Encl 4.1 Step 4.3). So it is reasonable to conclude the RO can select any two Nis for refueling.

Answer C Discussion

Incorrect: Plausible since only 1 SR NI is required to be operable while shutdown when fuel handling is not in progress. It would be reasonable to conclude that one is required any time while shutdown.

Answer D Discussion

CORRECT. Two Source Range Nis are required and they are selected by Reactor Engineering in accordance with OP/3/A/1502/007.

Basis for meeting the KA

Requires knowledge of the refueling process procedures.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT39 Q66

Development References

ILT39 Q66
FH-FHS Obj. 17
OP/3/A/1502/007

Student References Provided

GEN2.1 2.1.41 - GENERIC - Conduct of Operations
Conduct of Operations
Knowledge of the refueling process. (CFR: 41.2 / 41.10 / 43.6 / 45.13)

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 68 68 **B**

GEN2.2 2.2.22 - GENERIC - Equipment Control
Equipment Control
Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2)

Given the following Unit 1 condition:

- Reactor in MODE 1

Which ONE of the following is the MINIMUM Pressurizer level (inches) that would require declaring Tech Spec 3.4.9 (Pressurizer) LCO NOT met in accordance with PT/1/A/0600/001 (Periodic Instrument Surveillance)?

A. 240

B. 260

C. 285

D. 340

General Discussion

Answer A Discussion

Incorrect. Plausible since this value is below the TS required value of 285 therefore it is plausible to believe it to be an instrument corrected value. Also, 240 inches is the hi level alarm setpoint for the OAC alarm. Additional plausibility from the fact that this is a fairly common level value however it is the SG level required for natural circ when on EFDW.

Answer B Discussion

Correct. PT.600/01 corrects the TS required 285" for allowable instrument error and uses 260" as the threshold value.

Answer C Discussion

Incorrect. Plausible since this is the analytical value provided in Tech Spec 3.4.9 for maximum level.

Answer D Discussion

Incorrect. Plausible since this is a value associated with the pressurizer however this is the maximum Pzr level allowed for RCP restart with abnormal containment conditions.

Basis for meeting the KA

Requires knowledge of the conditions that would require the LCO requirments of TS 3.4.9 NOT met.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT44 Q59

Development References

Adm-ITS Obj R8
TS 3.4.9
PT/600/01

Student References Provided

GEN2.2 2.2.22 - GENERIC - Equipment Control
Equipment Control
Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2)

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination **QUESTION 69** 69 **C**

GEN2.2 2.2.39 - GENERIC - Equipment Control
Equipment Control
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:
Initial conditions:
- Reactor power = 100%

Current conditions:
- Keowee Emergency Start channels 1 and 2 inoperable

In accordance with Tech Spec 3.3.21 (EPSL Keowee Emergency Start Function) …

1) The MAXIMUM time allowed to declare both KHU's inoperable is __(1)__.

2) The Required Action associated with this inoperability is to energize __(2)__ Standby Bus(es) from a Lee Combustion Turbine via an isolated power path.

Which ONE of the following completes the statements above?

A. 1. immediately
2. at least ONE

B. 1. 1 hour
2. at least ONE

C. 1. immediately
2. BOTH

D. 1. 1 hour
2. BOTH

General Discussion

Answer A Discussion

Incorrect, First part is correct. Second part is incorrect. Plausible since it would be correct if only the KHU Underground powerpath were inoperable per TS 3.8.1 Condition D.

Answer B Discussion

Incorrect, First part is incorrect but plausible since 1 hour is a common completion time for significant inoperabilities. Additionally, allowing 1 hr to declare components inoperable is allowed in other TS's (Ex. 3.3.7) which adds to plausibility of the 1 hour completion time for declaring KHU's not operable. Also, when one KHU is declared inoperable, 1 hr is allowed to operability test the other KHU to determine if it is operable.Second part is incorrect. Plausible since it would be correct if only the KHU Underground powerpath were inoperable per TS 3.8.1 Condition D.

Answer C Discussion

Correct. Both Keowee units are required to be declared inoperable immediately per TS 3.3.21 Condition C. TS 3.8.1 Condition I requires energizing BOTH standby buses within 1 hr with both KHU's inoperable.

Answer D Discussion

Incorrect, First part is incorrect but plausible since 1 hour is a common completion time for significant inoperabilities. Additionally, allowing 1 hr to declare components inoperable is allowed in other TS's (Ex. 3.3.7) which adds to plausibility of the 1 hour completion time for declaring KHU's not operable. Also, when one KHU is declared inoperable, 1 hr is allowed to operability test the other KHU to determine if it is operable. Second part is correct.

Basis for meeting the KA

Requires knowledge of 1 hr or less actions required by TS 3.3.21 and 3.8.1.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	2009B NRC Exam Q70

Development References

2009B NRC Exam Q70
ADM-TSS Obj. R4
TS 3.3.21
TS 3.8.1

Student References Provided

GEN2.2 2.2.39 - GENERIC - Equipment Control

Equipment Control

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

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 Unit 3 conditions:

- 3A GWD gas tank release in progress
- Release is at 2/3 Station Limit

1) 3RIA-45 High and Alert setpoints will be set at __(1)__ the normal 1/3 Station Limit setpoint in accordance with PT/0/A/0230/001 (Radiation Monitor Check).

2) If 3RIA-45 High alarm setpoint is reached, the 3A GWD gas tank release __(2)__ automatically terminate.

Which ONE of the following completes the statements above?

A. 1. double
2. will

B. 1. double
2. will NOT

C. 1. half
2. will

D. 1. half
2. will NOT

General Discussion

Answer A Discussion

Incorrect.
First part is correct. Per PT/0/A/230/001, when performing a 2/3 station limit release, the releasing unit's RIA-45 setpoint is double the normal setpoint for 1/3 Station Limit.
Second part is incorrect. Plausible since RIA-45 high alarm will automatically terminate a RB Purge release on the respective unit.

Answer B Discussion

Correct.
First part is correct Per PT/0/A/230/001, the releasing unit's RIA-45 setpoint is double the normal setpoint for 1/3 Station Limit when performing a 2/3 station limit release.
Second part is correct. RIA-45 will not terminate the release. RIA-37/38 are the process monitors that are interlocked to terminate the release.

Answer C Discussion

Incorrect.
First part is incorrect. Plausible since PT/0/A/230/001 (Radiation Monitor Check) directs the setpoint for RIA-45 on the non-releasing unit to be set at half the 1/3 Station Limit value when release is at 2/3 station limit.
Second part is incorrect. Plausible since RIA-45 high alarm will automatically terminate a RB Purge release on the respective unit.

Answer D Discussion

Incorrect.
First part is incorrect. Plausible since PT/0/A/230/001 (Radiation Monitor Check) directs the setpoint for RIA-45 on the non-releasing unit to be set at half the 1/3 Station Limit value when release is at 2/3 station limit..
Second part is correct. RIA-45 will not terminate the release. RIA-37/38 are the process monitors that are interlocked to terminate the release.

Basis for meeting the KA

Question requires knowledge of the process for releasing at 2/3 the station limit.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	ILT39 Q70

Development References

ILT39 Q70
WE-GWD Obj. 07
OP/3/A/1104/018
PT/0/A/0230/001

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

GEN2.3 2.3.14 - GENERIC - Radiation Control
Radiation Control
Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (CFR: 41.12 / 43.4 / 45.10)

Given the following Unit 1 conditions:

- 36 gpm primary to secondary leak in 1A SG
- RCS cooldown in progress
- RCS temperature = 395°F decreasing
- The SRO has determined that current method of maintaining SCM is inadequate

1) The reason that the SGTR tab makes every effort to prevent a Reactor trip during shutdown is to minimize the chance of __(1)__ .

2) If an operator is dispatched to align Auxiliary Pressurizer Spray during the subsequent RCS cooldown and receives a dose alarm while performing this task he will __(2)__.

Which ONE of the following completes the statements above?

A. 1. lifting the MSRVs and causing a radiation release to the environment
2. complete the task while monitoring his dose

B. 1. lifting the MSRVs and causing a radiation release to the environment
2. immediately stop and leave the area

C. 1. RCS pressure transients and causing the Primary to Secondary leak rate to increase
2. complete the task while monitoring his dose

D. 1. RCS pressure transients and causing the Primary to Secondary leak rate to increase
2. immediately stop and leave the area

General Discussion

Answer A Discussion

Correct. The SGTR tab directs shutting down the plant as long as all available HPI can maintain Pzr level in order to prevent lifting the MSRV's would release to environment. Since emergency dose limits (EDLs) are in effect, the AO is expected to complete the task while monitoring his dose.

Answer B Discussion

Incorrect, First part is correct. The SGTR tab directs maintaining SCM low in order to reduce the dp between RCS and SG pressure, which in turn reduces the leak rate and contamination of the secondary plant. Second part is incorrect but plausible since it would be correct if EDLs were not in affect.

Answer C Discussion

Incorrect: First part is incorrect. Plausible since an RCS pressure transient could cause an increase in Primary to Secondary leakage however this is not the reason for guidance in the EOP for a controlled shutdown.
Second part is correct.

Answer D Discussion

Incorrect: First part is incorrect. Plausible since an RCS pressure transient could cause an increase in Primary to Secondary leakage however this is not the reason for guidance in the EOP for a controlled shutdown.
Second part is incorrect but plausible since it would be correct if EDLs were not in affect.

Basis for meeting the KA

Question requires knowledge of radiation hazards due to a SGTR in that it requires an understanding of the impact of lifting the MSRV's during a SGTR..

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	ILT42 Q70

Development References

ILT42 Q70
EAP-TCA Obj. R6
SGTR Tab Obj R1
EAP-SGTR

Student References Provided

GEN2.3 2.3.14 - GENERIC - Radiation Control

Radiation Control

Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (CFR: 41.12 / 43.4 / 45.10)

401-9 Comments:

Remarks/Status

GEN2.3 2.3.4 - GENERIC - Radiation Control
Radiation Control
Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12 / 43.4 / 45.10)

In accordance with PD-RP-ALL-0001 (Radiation Worker Responsibilities), which ONE of the following states…

1) the MAXIMUM annual Dose Limit (REM) allowed by the NRC?

2) the MAXIMUM lifetime Planned Special Exposure limit (REM) allowed by the NRC?

A. 1. 5
2. 25

B. 1. 5
2. 50

C. 1. 2
2. 25

D. 1. 2
2. 50

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72

A

General Discussion

Answer A Discussion

Correct. The maximum annual dose limit allowed by the NRC is 5 REM. The maximum lifetime planned special exposure limit allowed by the NRC is 5 times the annual limit (25 REM).

Answer B Discussion

Incorrect. First part is correct. Second part is plausible since 50 REM is the annual limit allowed by the NRC for SDE to skin and extremities and CDE to any tissue or organ except lens of the eye.

Answer C Discussion

Incorrect. First part is plausible since 2 REM is the annual dose limit allowed by Duke Energy. Second part is correct.

Answer D Discussion

Incorrect. First part is plausible since 2 REM is the annual dose limit allowed by Duke Energy. Second part is plausible since 50 REM is the annual limit allowed by the NRC for SDE to skin and extremities and CDE to any tissue or organ except lens of the eye.

Basis for meeting the KA

Requires knowledge of emergency worker radiation exposure limits and when they apply.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

2009B NRC Exam Q73
EAP-TCA Obj. R6
OMP 1-18
PD-RP-ALL-0001

Student References Provided

GEN2.3 2.3.4 - GENERIC - Radiation Control

Radiation Control

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

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 73 73 **B**

GEN2.4 2.4.16 - GENERIC - Emergency Procedures / Plan

Emergency Procedures / Plan

Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines. (CFR: 41.10 / 43.5 / 45.13)

Given the following Unit 1 conditions:
Initial conditions:

- Reactor power = 90% slowly decreasing
- SG Primary to Secondary leak rate = 6 gpm stable
- AP/31 (Primary to Secondary Leakage) in progress
- Unit shutdown in progress

Current conditions

- Reactor power = 60% slowly decreasing
- SG Primary to Secondary leak rate = 28 gpm slowly increasing

Which ONE of the following describes the actions required in accordance with plant procedures?

A. Continue unit shutdown using AP/31

B. Exit AP/31 and go directly to SGTR tab

C. Exit AP/31, perform IMA's, then go to SGTR tab

D. Perform AP/31 in parallel with performing the SGTR tab

ILT48 ONS SRO NRC Examination QUESTION 73

73

B

General Discussion

Answer A Discussion

Incorrect: Plausible since AP/31 would still be in effect if the leak rate were < 25 gpm and some AP's are self contained AP's that direct unit shutdowns and/or power decreases.

Answer B Discussion

Correct: There is an IAAT in AP/31 that directs going to the EOP if SG leak rate reaches 25 gpm. The entry conditions of the EOP with a SGTR > 25 gpm without a Reactor trip direct the operator to go directly to the SGTR tab.

Answer C Discussion

Incorrect: Plausible since exiting AP/31 and entering the EOP is correct and this would be the correct EOP path for most EOP entries however entering while on line with a SG tube rupture is a unique exception that requires going direclty to the SGTR tab.

Answer D Discussion

Incorrect: Plausible since most AP's are performed in parallel with the EOP when EOP entry is required.

Basis for meeting the KA

Requires knowledge of the hierarchy of the EOP versus AP/31.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	ILT40 Q74

Development References

ILT40 Q74
EAP-SA Obj. R21
AP/31
EOP

Student References Provided

GEN2.4 2.4.16 - GENERIC - Emergency Procedures / Plan

Emergency Procedures / Plan

Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines. (CFR: 41.10 / 43.5 / 45.13)

401-9 Comments:

Remarks/Status

GEN2.4 2.4.45 - GENERIC - Emergency Procedures / Plan
Emergency Procedures / Plan
Ability to prioritize and interpret the significance of each annunciator or alarm. (CFR: 41.10 / 43.5 / 45.3 / 45.12)

Given the following Unit 1 conditions:

- Reactor tripped from 100% power
- The following Statalarms actuate:
  - 1SA-1/C-11 (ES Channel 7 Trip)
  - 1SA-1/D-11 (ES Channel 8 Trip)
  - 1SA-2/C-4 (RC Pressurizer Level Emerg High/Low)
  - 1SA-2/C-8 (AFIS Header A Initiated)
  - 1SA-2/D-5 (HP LDST Level Interlock Initiated)
  - 1SA-8/A-3 (FDWPT A Trip)
  - 1SA-8/A-6 (FDWPT B Trip)

Which ONE of the following emergency procedures has the highest priority?

A. EOP Enclosure 5.1 (ES Actuation)

B. Rule 5 (Main Steam Line Break)

C. EOP Enclosure 5.5 (Pzr and LDST Level Control)

D. Rule 3 (Loss of Main or Emergency Feedwater)

General Discussion

Answer A Discussion

Incorrect: Plausible in that ES Actuation of all 8 Channels has most likely occurred

Answer B Discussion

CORRECT: AFIS actuation tells the operator that MSLB has occurred which is the highest priority due to the overcooling it can cause; in fact EHT has caused all the other SA conditions

Answer C Discussion

Incorrect: Plausible in that overcooling is causing the RCS inventory to contract to the point of possibly emptying the Pzr and LDST; however the overcooling would be stopped by Rule 5.

Answer D Discussion

Incorrect: Plausible in that AFIS has caused the MFWPs to trip; a loss of both MFW and EFW would be a higher priority than Rule 5 but there is no reason to assume that EFW pumps are unavailable.

Basis for meeting the KA

Requires the ability to interpret the significance of statalarms, to diagnose the event, and to prioritize EOP rules and enclosures

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	2009 NRC Exam Q74

Development References

2009 NRC Exam Q74
Admin-OMP Obj R10, R52
OMP 1-18

Student References Provided

GEN2.4 2.4.45 - GENERIC - Emergency Procedures / Plan

Emergency Procedures / Plan

Ability to prioritize and interpret the significance of each annunciator or alarm. (CFR: 41.10 / 43.5 / 45.3 / 45.12)

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 75 75 **D**

GEN2.4 2.4.49 - GENERIC - Emergency Procedures / Plan

Emergency Procedures / Plan

Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (CFR: 41.10 / 43.2 / 45.6)

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 = 3% slowly decreasing

Which ONE of the following describes the NEXT action required in accordance with EOP Immediate Manual Actions?

A. Perform Rule 1

B. OPEN 1HP-24 and 1HP-25

C. Verify RCP seal injection available

D. Depress the Turbine TRIP pushbutton

ILT48 ONS SRO NRC Examination QUESTION 75

75

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 aligning emergency boration 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 Immediate Manual Actions from memory.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	ILT40 Q75

Development References

ILT40 Q75
EAP-SA R24
EOP IMAs

Student References Provided

GEN2.4 2.4.49 - GENERIC - Emergency Procedures / Plan

Emergency Procedures / Plan

Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (CFR: 41.10 / 43.2 / 45.6)

401-9 Comments:

Remarks/Status

APE022 2.4.20 - Loss of Reactor Coolant Makeup

APE022 GENERIC

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

Given the following Unit 1 conditions:

- RCS Tave = 480°F stable
- 1A HPI pump tripped
- 1B HPI pump failed to start and can NOT be started manually
- In accordance with AP/14 (Loss of Normal HPI Makeup and/or RCP Seal Injection), BOTH the SSF RC Makeup Pump and the 1C HPI Pump are aligned to supply RCP seal injection

In accordance with AP/14…

1) __(1)__ will be utilized to secure one of the sources of RCP seal injection.

2) An RCS Heatup of $\leq$ 10ºF __(2)__ allowed as part of the strategy to maintain Pressurizer level $\geq$ 100 inches

A. 1. Encl. 5.1 (SSF RC Makeup)
   2. is

B. 1. Encl. 5.1 (SSF RC Makeup)
   2. is NOT

C. 1. Encl. 5.2 (Emergency Alignment of 1C HPI Pump for Normal Makeup)
   2. is

D. 1. Encl. 5.2 (Emergency Alignment of 1C HPI Pump for Normal Makeup)
   2. is NOT

ILT48 ONS SRO NRC Examination QUESTION 76

76

A

General Discussion

Answer A Discussion

Correct. Even though the preferred method of providing Seal Injection is with the C HPI pump, both paths are pursued and then the RC Makeup pump is secured when and if the C HPI pump is successfully aligned and operating. There is a NOTE in the AP that allows heating up as much as 10 degrees to help maintain Pzr level no less than 100 inches as long as the Rx is shutdown.

Answer B Discussion

Incorrect. First part is correct. Second part is plausible since it would not be allowed if the Reactor were not shut down. Also plausible to believe that heating up would not be allowed to control Pzr level since there is so much focus on controlling RCS heatup when it is occurring.

Answer C Discussion

Incorrect. First part is plausible since Encl. 5.1 (SSF RC Makeup) is initiated prior to Encl. 5.2 and only secured when the 1C HPI Pump is successfully aligned per Encl. 5.2. The operator could reason that since Encl. 5.1 is initiated first, that it is the preferred method to supply seal injection. Second part is correct.

Answer D Discussion

Incorrect.First part is plausible since Encl. 5.1 (SSF RC Makeup) is initiated prior to Encl. 5.2 and only secured when the 1C HPI Pump is successfully aligned per Encl. 5.2. The operator could reason that since Encl. 5.1 is initiated first, that it is the preferred method to supply seal injection.
Second part is plausible since it would not be allowed if the Reactor were not shut down. Also plausible to believe that heating up would not be allowed to control Pzr level since there is so much focus on controlling RCS heatup when it is occurring.

Basis for meeting the KA

Requires knowledge of the operational implication of a Note at step 29 of AP/14 allowing RCS heatup only if the Rx is shutdown.

Basis for Hi Cog

Basis for SRO only

This question requires assessing plant conditions and determining a section of a procedure to be utilized based on that assessment. Both parts of this question meet that SRO guideline criteria since both determine steps or sections to be performed in the AP which are based solely on plant conditions. 10 CFR 55.43(b)(5).

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References	Student References Provided
EAP/APG R9 AP/14	

APE022 2.4.20 - Loss of Reactor Coolant Makeup
APE022 GENERIC
Knowledge of the operational implications of EOP warnings, cautions, and notes. (CFR: 41.10 / 43.5 / 45.13)

401-9 Comments:	Remarks/Status

APE027 AA2.04 - Pressurizer Pressure Control System (PZR PCS) Malfunction
Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: (CFR: 43.5 / 45.13)
Tech-Spec limits for RCS pressure ....

Given the following Unit 1 conditions:

- Reactor in MODE 5
- ALL LTOP requirements established in accordance with Tech Spec 3.4.12 (Low Temperature Overpressure Protection (LTOP) System)

In accordance with Tech Spec 3.4.12…

1) The 1RC-66 setpoint is __(1)__ psig.

2) A dedicated LTOP operator __(2)__ allowed as a substitute for an inoperable PORV to prevent exceeding RCS brittle fracture pressure limits.

Which ONE of the following completes the statements above?

A. 1) 535
2) is

B. 1) 535
2) is NOT

C. 1) 550
2) is

D. 1) 550
2) is NOT

ILT48 ONS SRO NRC Examination QUESTION 77

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B

General Discussion

Answer A Discussion

Incorrect: First part is correct. Second part is incorrect. Plausible since there are 2 trains of LTOP. The trains are the PORV and Administrative Controls that limit the rate of RCS pressure increase. A dedicated LTOP operator is allowed as a substitute for one of the trains (Admin. Controls), but not the other (PORV).

Answer B Discussion

Correct: The 1RC-66 setpoint per TS 3.4.12 is 535 psig. A dedicated LTOP operator is not allowed as a substitute for an inoperable PORV.

Answer C Discussion

Incorrect: First part is incorrect. Plausible since 550 psig is the setpoint for LPI actuation from ES 3/4 and is also the main steam setpoint for Automatic Feedwater Isolation (AFIS).
Second part is incorrect. Plausible since there are 2 trains of LTOP. The trains are the PORV and Administrative Controls that limit the rate of RCS pressure increase. A dedicated LTOP operator is allowed as a substitute for one of the trains (Admin. Controls), but not the other (PORV).

Answer D Discussion

Incorrect: First part is incorrect. Plausible since 550 psig is the setpoint for LPI actuation from ES 3/4 and is also the main steam setpoint for Automatic Feedwater Isolation (AFIS). Second part is correct.

Basis for meeting the KA

Requires knowledge of the LTOP LCO as it relates to the PORV,
which is part of the Pressurizer pressure control system. The malfunction portion of
the KA is satisfied by the failure of the PORV to open. With the failure having occurred the operator is being asked to interpret the failure as it relates to exceeding the Tech Spec RCS pressure limits that would result in RCS brittle fracture.

Basis for Hi Cog

Basis for SRO only

In accordance with Clarification Guidance for SRO-only Questions:
This question requires knowledge from the basis of TS 3.4.12 and the Safety Limits that
is not systems knowledge. 10CFR 55.43(b)(2).
It cannot be answered by knowing 1 hr or less TS/TRM Action
It cannot be answered solely with "above the line" information.
It cannot be answered solely by knowing Safety Limits

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References

ADMIN-TSS obj R5
TS 3.4.12 and basis

Student References Provided

APE027 AA2.04 - Pressurizer Pressure Control System (PZR PCS) Malfunction
Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: (CFR: 43.5 / 45.13)
Tech-Spec limits for RCS pressure ....

401-9 Comments:

Remarks/Status

Preview

EPE038 EA2.01 - Steam Generator Tube Rupture (SGTR)
Ability to determine or interpret the following as they apply to a SGTR : (CFR 43.5 / 45.13)
When to isolate one or more S/Gs ....

Given the following Unit 1 conditions:
Time = 1000:

- MSLB on 1B SG outside containment and can NOT be steamed
- 1A S/G developed a SGTR
- EHT tab is complete
- EOP SGTR Tab in progress
- 1A S/G level = 288” XSUR slowly increasing
- SCM = 10°F stable

Time = 1040:

- 1A S/G has reached the level where water can enter the main steam lines
- Rule 4 has been initiated and the PORV has been opened
- Core SCM = 0°F

1) At Time = 1000, the SGTR tab __(1)__ direct performance of Encl. 5.22 (SG Blowdown)

2) At Time = 1040, the SGTR tab will direct initiation of the __(2)__ tab.

Which ONE of the following completes the statements above?

A. 1. does
2. HPI CD

B. 1. does
2. LOCA CD

C. 1. does NOT
2. HPI CD

D. 1. does NOT
2. LOCA CD

General Discussion

Answer A Discussion

1st part is correct because per the STGR tab (step 227) Encl 5.22 is performed if any SG with a tube rupture rises > 285".

2nd part is correct. Per step 232 and 233 in the SGTR tab, stop steaming the affected SG and if the remaining SG is not available for steam (RNO) Perform Rule 4 and GO TO the HPI CD tab.

Answer B Discussion

1st part is correct.

2nd part is incorrect because the SGTR tab directs you to the HPI CD tab. It is plausible because if HPI FC was not in progress, it would be correct. LOSCM is the higher priority tab (normally).

Answer C Discussion

1st part is incorrect. The STGR tab (step 227): Encl 5.22 is performed if any SG with a tube rupture rises > 285". Plausible since 285" is a rather arbitrary number and there are actions in the EOP that occure at 290 psi RCS pressure therefore confusing the numbers would be plausible.

2nd part is correct. Per step 232 and 233 in the SGTR tab, stop steaming the affected SG and if the remaining SG is not available for steam (RNO) Perform Rule 4 and GO TO the HPI CD tab.

Answer D Discussion

1st part is incorrect. The STGR tab (step 227): Encl 5.22 is performed if any SG with a tube rupture rises > 285". Plausible since 285" is a rather arbitrary number and there are actions in the EOP that occure at 290 psi RCS pressure therefore confusing the numbers would be plausible.

2nd part is incorrect because the SGTR tab directs you to the HPI CD tab. It is plausible because if HPI FC was not in progress, it would be correct. LOSCM is the higher priority tab (normally).

Basis for meeting the KA

The question requires knowledge of the isolation requirements of a SG since knowing when steaming of an isolated SG is required demonstrates the ability to determin if the SG should be isolated or not.

Basis for Hi Cog

Basis for SRO only

This question is SRO ONLY because it meets. 10 CFR 55.43(b)(5)

It requires assessing plant conditions and determining when a section of the EOP should be performed.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	MODIFIED	ILT43 Q78

Development References

ILT43 Q78
SGTR Tab
EAP SGTR Obj R 16, R26

Student References Provided

EPE038 EA2.01 - Steam Generator Tube Rupture (SGTR)
Ability to determine or interpret the following as they apply to a SGTR : (CFR 43.5 / 45.13)
When to isolate one or more S/Gs ....

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 79 79 **C**

APE057 2.4.11 - Loss of Vital AC Electrical Instrument Bus
APE057 GENERIC
Knowledge of abnormal condition procedures. (CFR: 41.10 / 43.5 / 45.13)

Given the following Unit 1 conditions:

- Reactor power = 50% stable
- 1KI and 1KU panelboards are de-energized

Which ONE of the following states the correct procedure to use to maintain Steam Generator pressure and RCS temperature?

A. AP/23 (Loss of ICS Power)

B. AP/25 (Standby Shutdown Facility)

C. EOP Enclosure 5.24 (Operation of the ADVs)

D. EOP Enclosure 5.42 (Alignment of EFM Pump to Feed SGs)

ILT48 ONS SRO NRC Examination QUESTION 79

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C

General Discussion

Answer A Discussion

Incorrect. Plausible since the failures will meet the entry conditions for AP/23 and if the other essential inverter (KX) were lost instead of one of these two it would be a correct answer if it occurred in MODE 3 or 4.

Answer B Discussion

Incorrect. Plausible since this is a method used by the EOP to maintain stable SG pressure and RCS temperature under other circumstances and with both KI and KU being lost with the unit at power a Rx trip would have occurred therefore the entry condtions to the EOP would be met.

Answer C Discussion

Correct. With both KI and KU lost the TBV's are unavailable and section 4A of AP/23 will direct using EOP Encl 5.24.

Answer D Discussion

Incorrect. Plausible since Enclosure 5.42 is utilized to feed the SGs under certain loss of power scenarios.

Basis for meeting the KA

CE allowed use of Essential Inverters since Oconee does not have AP's realted to loss of vital inverters. This question requires knowledge of Abnormal Procedures realted to a loss of essential inverters.

Basis for Hi Cog

Basis for SRO only

Question requires selection of procedure and cannot be answered based solely on entry conditions. 10 CFR 55.43(b)(5).

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References

EAP-APG obj R9
EOP IMA's and SA's
AP/23

Student References Provided

APE057 2.4.11 - Loss of Vital AC Electrical Instrument Bus
APE057 GENERIC
Knowledge of abnormal condition procedures. (CFR: 41.10 / 43.5 / 45.13)

401-9 Comments:

Remarks/Status

Preview

APE065 2.4.41 - Loss of Instrument Air
APE065 GENERIC
Knowledge of the emergency action level thresholds and classifications. (CFR: 41.10 / 43.5 / 45.11)

Given the following Unit 1 conditions:
Time = 1200:
- Reactor trip from 100%
- AP/22 (Loss of Instrument Air) in progress
- Instrument Air (IA) and Auxiliary Instrument Air (AIA) pressure lost to the 1A Turbine Bypass Valves
- 1A SG pressure 935 psig stable
- Maintaining 1A SG pressure with Atmospheric Dump Valves (ADVs) in accordance with EOP Subsequent Actions

Time = 1210:
- 1A SG Tube Leak = 15 gpm stable

Time = 1230:
- 1A SG Tube Rupture = 180 gpm stable
- EOP SGTR in progress

1) At time = 1210, the emergency classification is __(1)__.

2) At time = 1230, the emergency classification is __(2)__.

Which ONE of the following completes the statements above?

REFERENCE PROVIDED

A. 1. NONE
   2. ALERT

B. 1. NONE
   2. SITE AREA EMERGENCY

C. 1. UNUSUAL EVENT
   2. ALERT

D. 1. UNUSUAL EVENT
   2. SITE AREA EMERGENCY

General Discussion

Answer A Discussion

Incorrect: First part is incorrect. Plausible since it would be correct if the ADVs were not being used to steam 1A SG. Second part is incorrect. Plausible since it would be correct if the ADVs were not being used to steam 1A SG.

Answer B Discussion

Incorrect: First part is incorrect. Plausible since it would be correct if the ADVs were not being used due to the loss of IA and AIA to 1A TBV. Second part is correct.

Answer C Discussion

Incorrect. First part is correct. Second part is incorrect. Plausible since it would be correct if the ADVs were not being used to steam 1A SG.

Answer D Discussion

CORRECT. The classification at 1210 is Unusual Event. This is based on receiving 3 points under Containment Barriers (Loss) on the Fission Product Barrier Matrix for "Failure of secondary side of SG results in a direct opening to the environment with SG Tube Leak >/= 10 gpm in the same SG". Unusual Event is 1 to 3 points.
The classification at 1230 is Site Area Emergency. This is based on the 3 points described above plus 4 points received under RCS Barriers (Potential Loss) for "SGTR >/= 160 gpm" for a total of 7 points. SAE is 7 to 10 points.

Basis for meeting the KA

Question requires the candidate to evaluate plant conditions associated with loss of Instrument Air and SGTR and make emergency classifications based on their evaluation.

Basis for Hi Cog

Basis for SRO only

Question requires the applicant to evaluate plant conditions and make emergency classifications based on their evaluation.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References
EAP-SEP Obj. R12 RP/0/A/1000/001 E-Plan Basis Doc.

Student References Provided
RP/0/A/1000/001

APE065 2.4.41 - Loss of Instrument Air
APE065 GENERIC
Knowledge of the emergency action level thresholds and classifications. (CFR: 41.10 / 43.5 / 45.11)

401-9 Comments:

Remarks/Status

APE077 AA2.07 - Generator Voltage and Electric Grid Disturbances

Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: (CFR: 41.5 and 43.5 / 45.5, 45.7, and 45.8)

Operational status of engineered safety features..........................

Given the following Unit 1 conditions:
Time = 1200

- Reactor power = 100% stable
- AFIS bypassed for maintenance
- SA-16/C-1 (230 KV Swyd Isolate ES Permit) actuated
- 230 KV Yellow Bus voltage = 224.2 KV increasing

Time = 1201

- AP/34 (Degraded Grid) in progress
- Main Turbine trips due to low grid frequency
- 230 KV Yellow Bus voltage = 226.8 KV increasing
- RCS pressure = 1245 psig rapidly decreasing
- RB pressure = 11.4 psig rapidly increasing

Engineered Safeguards systems...

1) will be energized from __(1)__.

2) __(2)__ sufficient to maintain Reactor Building pressure within design limits.

Which ONE of the following completes the statements above?

A. 1. CT-1
2. are

B. 1. CT-1
2. are NOT

C. 1. CT-4
2. are

D. 1. CT-4
2. are NOT

General Discussion

Answer A Discussion

Incorrect. First part is plausible since it would be correct if there had not been a switchyard isolation. Second part is correct.

Answer B Discussion

Incorrect. First part is plausible since it would be correct if there had not been a switchyard isolation.
Second part is plausible since it would be correct if this were a MSLB since AFIS is bypassed. The basis of TS 3.3.11 (AFIS) explains that for a Main Steam Line break (which is what is occuring in this question)

Answer C Discussion

Correct.
Grid voltage is low and if it stays less than 227,468 for greater than 9 seconds then an ES 1 or 2 actuation on any unit will cause a swyd isolation to occur. In the current conditions swyd voltage is still low along with low RCS pressure which causes a swyd isolation to occur due to ES 1 and 2 actuation. A swyd isolation concurrent with an ES actuation will result in power to the MFBs coming from a Keowee unit via the underground and CT-4.
The basis of TS 3.6.5 (Reactor Building Spray and Reactor Building Cooling) explains that for a LOCA (which is what is occuring in this question) the highest peak containment pressure is 57.75 psig

Answer D Discussion

Incorrect.
First part is correct.
Second part is plausible since it would be correct if this were a MSLB since AFIS is bypassed. The basis of TS 3.3.11 (AFIS) explains that for a Main Steam Line break (which is what is occuring in this question)

Basis for meeting the KA

Requires knowledge of the operational status of ES components following a grid distrubance. Since without the grid distrubance, the supply to ES components would be CT-1 this question meets the KA. It is met at the SRO level since the LOCA is initiated by a turbine trip that is initiated by the grid distrubance.

Basis for Hi Cog

Basis for SRO only

This question is SRO only in that it requires knowledge of design basis of plant that is not systems knowledge. This information is incorporated into the basis of Tech Spec 3.6.5 and 3.3.11and is therefore basis knowledge and as allowed by SRO clarification guidance it becomes an SRO level question. 10 CFR 55.43(b)(2)

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	MODIFIED	ILT39 Q16

Development References

ILT39 Q16
TS 3.3.11` basis
TS 3.6.5 basis
EL-PSL Obj 11

Student References Provided

APE077 AA2.07 - Generator Voltage and Electric Grid Disturbances

Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: (CFR: 41.5 and 43.5 / 45.5, 45.7, and 45.8)

Operational status of engineered safety features...........................

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 82 82 **C**

APE003 AA2.02 - Dropped Control Rod

Ability to determine and interpret the following as they apply to the Dropped Control Rod: (CFR: 43.5 / 45.13)

Signal inputs to rod control system ....

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%
- Control Rod 3 in Group 7 indicates 0% withdrawn
- AP/01 (Unit Runback) initiated

Current conditions:

- Reactor power 70% decreasing

1) With ICS in AUTO, the Regulating Rods will NOT respond to an "out" command until Reactor power is less than a MAXIMUM of __(1)__ percent.

2) The basis for the above limit on Reactor power is to maintain __(2)__ within design limits.

Which ONE of the following completes the statements above?

A. 1. 55
   2. Linear Heat Rate

B. 1. 55
   2. Shutdown Margin

C. 1. 60
   2. Linear Heat Rate

D. 1. 60
   2. Shutdown Margin

ILT48 ONS SRO NRC Examination QUESTION 82

82

C

General Discussion

Answer A Discussion

Incorrect. First part is incorrect. Plausible since it is the value for the Asymmetric Rod Runback. Second part is correct.

Answer B Discussion

Incorrect. First part is incorrect. Plausible since it is the value for the Asymmetric Rod Runback. Second part is incorrect. Plausible since SDM at power is verified using curves in the COLR which utilize reactor power and control rod position. Additionally, the operator utilizes a different curve when operating with a dropped rod.

Answer C Discussion

CORRECT. With an Asymmetric/dropped rod, a Regulating Rod "out inhibit" is established if in manual OR if in AUTO with NI power > 60%. The basis for the limit on reactor power is to ensure that local Linear Heat Rate increases, due to the misaligned rod, will not cause the core design criteria to be exceeded.

Answer D Discussion

Incorrect. First part is correct. Second part is incorrect. Plausible since SDM at power is verified using curves in the COLR which utilize reactor power and control rod position. Additionally, the operator utilizes a different curve when operating with a dropped rod.

Basis for meeting the KA

Question requires the ability to determine the power level, above which, the rod control system will not respond to an "out" command and the basis for the limit.

Basis for Hi Cog

Basis for SRO only

The question cannot be answered solely by knowing 1 hour or less actions, "above the line" information, or TS Safety Limits. It requires knowledge of TS Bases. 10 CFR 55.43(b)(2)

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	NEW	

Development References

IC-CRI Obj. 11
ADM-TSS Obj. R5
TS 3.1.4 Bases

Student References Provided

APE003 AA2.02 - Dropped Control Rod

Ability to determine and interpret the following as they apply to the Dropped Control Rod: (CFR: 43.5 / 45.13)

Signal inputs to rod control system ....

401-9 Comments:

Remarks/Status

APE024 AA2.06 - Emergency Boration

Ability to determine and interpret the following as they apply to the Emergency Boration: (CFR: 43.5 / 45.13)

When boron dilution is taking place ....

Given the following Unit 1 conditions:

- Reactor power = 75% stable
- Group 7 rod position = 72% withdrawn stable
- Makeup to LDST initiated

1) If the LDST makeup were diluting boron concentration, Group 7 rods would be __(1)__.

2) If the boron dilution results in NOT meeting the Regulating rod position limits of Tech Spec 3.2.1 (Regulating Rod Position Limits), Tech Spec requires that the makeup source for boration be either the __(2)__ or the Concentrated Boric Acid Storage Tank.

A. 1. withdrawing
   2. BWST

B. 1. withdrawing
   2. “A” BHUT

C. 1. inserting
   2. BWST

D. 1. inserting
   2. “A” BHUT

ILT48 ONS SRO NRC Examination QUESTION 83 83 **C**

General Discussion

Answer A Discussion

Incorrect. First part is plausible as it is a common error to confuse rod motion direction with boron changes vs temperature changes. Second part is correct.

Answer B Discussion

Incorrect. First part is plausible as it is a common error to confuse rod motion direction with boron changes vs temperature changes.
Second part is plausible since Tech Specs requires that a minimum BWST level of 46 feet be maintained as part of ECCS support therefore it is plausible to believe that TS would not direct using BWST if there were other options in order to preserve its design function availability. Since A BHUT is maintained at a higher boron concentration than RCS it is a plausible alternative to the BWST since it would result in restoring SDM. Additional plausibility is derived from the basis requirements of TS 3.1.8 which allow boration from the "best source available for unit conditions" when SDM requirements are not met therefore this would be a correct answer if the SDM requirements were not met during Physics Testing.

Answer C Discussion

Correct. If boron concentration were decreasing then control rods would be inserting to offset the decrease in boron. The basis of TS 3.2.1 requires that the boration be done in the manner described in the basis of TS 3.1.1 and that states that the boration should be from a highly borated source such as CBAST or the BWST.

Answer D Discussion

Incorrect. First part is correct.
Second part is plausible since Tech Specs requires that a minimum BWST level of 46 feet be maintained as part of ECCS support therefore it is plausible to believe that TS would not direct using BWST if there were other options in order to preserve its design function availability. Since A BHUT is maintained at a higher boron concentration than RCS it is a plausible alternative to the BWST since it would result in restoring SDM. Additional plausibility is derived from the basis requirements of TS 3.1.8 which allow boration from the "best source available for unit conditions" when SDM requirements are not met therefore this would be a correct answer if the SDM requirements were not met during Physics Testing.

Basis for meeting the KA

This question meets the KA in that it requires being able to determine when a boron dilution is taking place by observing the resulting control rod motion. Additionally, at the SRO level the ability to "interpret" the boron dilution would include the ability to mitigate the event and to answer that part of this question requires TS basis knowledge.

Basis for Hi Cog

Basis for SRO only

In accordance with Clarification Guidance for SRO-only Questions:
This question requires knowledge from the basis of TS 3.2.1 and 3.1.1 and
is not systems knowledge. 10 CFR 55.43(b)(2)
It cannot be answered by knowing 1 hr or less TS/TRM Action
It cannot be answered solely with "above the line" information.
It cannot be answered solely by knowing Safety Limits

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	MODIFIED	ILT46 Q96

Development References

ILT46 Q96
TS 3.2.1 basisTS
TS 3.1.1 basis
TS 3.1.8 basis

Student References Provided

APE024 AA2.06 - Emergency Boration
Ability to determine and interpret the following as they apply to the Emergency Boration: (CFR: 43.5 / 45.13)
When boron dilution is taking place ....

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 84 84 **C**

BWA05 2.2.44 - Emergency Diesel Actuation

BWA05 GENERIC

Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. (CFR: 41.5 / 43.5 / 45.12)

Given the following Unit 1 conditions:
Initial conditions:

- Reactor power = 100%
- ACB-3 closed

Current conditions:

- 2SA-17/A-1 (GEN #1 EMERG. LOCKOUT) actuated
- 1TC, 1TD, and 1TE 4160V switchgear are Locked Out
- Blackout tab in progress
- EOP Enclosure 5.38 (Restoration of Power) has been initiated

1) Main Feeder Buses will be energized by __(1)__.

2) In accordance with the Blackout tab, once the Main Feeder Buses are energized the CRS will be directed to __(2)__.

Which ONE of the following completes the statements above?

A. 1. CT-5
2. continue in the Blackout Tab

B. 1. CT-5
2. transfer back to Subsequent Actions Tab

C. 1. KHU-2
2. continue in the Blackout Tab

D. 1. KHU-2
2. transfer back to Subsequent Actions Tab

General Discussion

Answer A Discussion

Incorrect. First part is plausible since it would be correct if neither of the KHU's were available. Second part is correct.

Answer B Discussion

Incorrect. First part is plausible since it would be correct if neither of the KHU's were available. Second part is plausible since it would be correct if any one of the three 4160v buses were energized by the MFB.

Answer C Discussion

Correct. Guidance in Encl. 5.38 directs closing the underground feeder (ACB-4) for the operating KHU to see if that will get the MFBs energized before going to CT-5. The transfer back to Subsequent Actions requires at least one of the 4160v switchgear buses to be energized even after the MFB's are energized.

Answer D Discussion

Incorrect. First part is correct, Second part is plausible since it would be correct if any one of the three 4160v buses were energized by the MFB.

Basis for meeting the KA

Requires the ability to interpret control room indications to determine the status of available power sources to understand which power source will be aligned to energize the Main Feeder Buses (MFBs). Also requires an understanding of operator actions to restore power such that even though power is restored to the MFBs, realize that the 4160V Switchgear is still de-energized and based on these system conditions, select the appropriate EOP section which is to continue in the Blackout Tab.

Basis for Hi Cog

Basis for SRO only

Requires assessing operator actions and plant conditions to determine a procedure or section of a procedure with which to proceed. 10 CFR 55.43(b)(5).

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	BANK	ILT45 Q85

Development References

ILT45 Q85
EAP-BO Obj. R1 & R6
EOP Encl. 5.38
Blackout Tab

Student References Provided

BWA05 2.2.44 - Emergency Diesel Actuation

BWA05 GENERIC

Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. (CFR: 41.5 / 43.5 / 45.12)

401-9 Comments:

Remarks/Status

BWA07 2.4.21 - Flooding

BWA07 GENERIC

Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (CFR: 41.7 / 43.5

Given the following Unit 1 conditions:

Time = 1200:

- Notified by Unit 2 that 2SA-18/A-11 (TURBINE BSMT WATER EMERGENCY HIGH LEVEL) in alarm
- Reactor tripped from 100%
- EOP Turbine Building Flood (TBF) tab initiated

Time = 1300:

- Core SCM = 0°F

In accordance with the EOP TBF tab...

1) At time = 1200, initiate feeding SGs with Main and Emergency Feedwater while maintaining a MINIMUM Tave of __(1)__ºF.

2) At time = 1300, performance of Rule 4 (Initiation of HPI Forced Cooling) __(2)__ required.

Which ONE of the following completes the statements above?

A. 1. 550
2. is

B. 1. 532
2. is

C. 1. 550
2. is NOT

D. 1. 532
2. is NOT

ILT48 ONS SRO NRC Examination QUESTION 85

85 **B**

General Discussion

Answer A Discussion

First part is incorrect. Plausible since a minimum of 550 degrees Tc (550 - 555) is maintained per EOP Rule 7 during an SSF Event in Mode 1 or 2 with TDEFDWP or alternate unit providing SG Feed and the TBF tab will utilize the SSF to feed the SGs if no main or emergency FDW is available.

Second part is correct. Step 20 in the TBF tab states that IAAT RCS heats to the point of losing core SCM, THEN GO TO STEP 88 (Perform Rule 4).

Answer B Discussion

First part is correct. The EOP TBF tab initiates feeding both SGs to 95% Operating Range with Main and Emergency Feedwater. If initial Tave >/= 532 degrees, Tave is maintained >/= 532 degrees during the SG fill.

Second part is correct. If Core SCM is lost due to heat up, the TBF tab directs performance of Rule 4.

Answer C Discussion

First part is incorrect. Plausible since a minimum of 550 degrees Tc (550 - 555) is maintained per EOP Rule 7 during an SSF Event in Mode 1 or 2 with TDEFDWP or alternate unit providing SG Feed and the TBF tab will utilize the SSF to feed the SGs if no main or emergency FDW is available.

Second part is incorrect. Plausible since the EHT and SGTR tabs direct transfer to the LOSCM tab when any SCM = 0 and HPI Forced Cooling is not in progress.

Answer D Discussion

First part is correct. The EOP TBF tab initiates feeding both SGs to 95% Operating Range with Main and Emergency Feedwater. If initial Tave >/= 532 degrees, Tave is maintained >/= 532 degrees during the SG fill.

Second part is incorrect. Plausible since the EHT and SGTR tabs direct transfer to the LOSCM tab when any SCM = 0 and HPI Forced Cooling is not in progress.

Basis for meeting the KA

Question requires knowledge of a parameter (Core SCM) and logic (Core SCM lost due to heat up) during a Turbine Building Flood to assess the status of core cooling and heat removal and transfer to a different section of the EOP TBF tab to perform Rule 4 (Initiation of HPI Forced Cooling).

Basis for Hi Cog

Basis for SRO only

Per ES-401 Attachment 2 Clarification Guidance for SRO Only Questions, the candidate must assess plant conditions and then select a procedure or section of a procedure with which to proceed. 10 CFR 55.43(b)(5).

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References
EAP-TBF Obj. R4 / R5 EOP TBF

Student References Provided

BWA07 2.4.21 - Flooding

BWA07 GENERIC

Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (CFR: 41.7 / 43.5

401-9 Comments:	Remarks/Status
	Preview

SYS005 2.2.25 - Residual Heat Removal System (RHRS)

SYS005 GENERIC

Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. (CFR: 41.5 / 41.7 / 43.2)

Given the following Unit 1 conditions:

Time = 1215

- Unit shutdown in progress
- 1A1 RCP in operation
- Tcold = 220°F slowly decreasing
- RCS Pressure 325 psig slowly decreasing
- LPI Cooler outlet temperature = 110°F stable

Time = 1230

- Tcold = 200°F slowly decreasing
- RCS Pressure 275 psig stable
- LPI Cooler outlet temperature = 110°F stable
- 1A LPI Pump is started

Time = 1240

- Tcold = 197°F stable
- 1A1 RCP secured

Time = 1245

- LPI Cooler outlet temperature = 180°F stable

1) The RCS cooldown rate __(1)__ violate the maximum cooldown rate allowed per Tech Specs?

2) When the 1A LPI pump is started at Time = 1230, the temperature transient that results from the difference in LPI Cooler Outlet temperature and Tcold __(2)__ outside the bounds of the Reactor Vessel stress analysis?

Which ONE of the following completes the statements above?

A. 1. does
2. is

B. 1. does
2. is NOT

C. 1. does NOT
2. is

D. 1. does NOT
2. is NOT

General Discussion

Answer A Discussion

Incorrect. First part is correct. Second part is plausible since TS specifies LPI cooler Outlet temperature as RCS temperature when on LPI (with NO RCP's operating). Since the delta temp between cooler outlet and Tcold is in excess of the allowed cooldown rate for this RCS temperature it would be plausible to believe that the cooldown limits would be exceeded.

Answer B Discussion

Correct. RCS temperature at 1215 is 220 degrees since there is an RCP in operation. At 1245 RCS temp is 180 degrees since there is only an LPI pump in operation. The cooldown would be 40 degrees. The TS limit is 25 degrees in any 1/2 hour at this RCS temperature therefore the TS limit has been exceeded. The TS basis for Cooldown Rates (TS 3.4.3) specifically addresses the RV issue as follows:
An analysis examined the effects of initiating flow through a previously idle LPI train (i.e. either placing a train of LPI in operation or swapping from one train to the other) when none of the RC pumps are operating. This analysis has determined that the brief temperature excursion caused by the fluid initially in the idle LPI train can be accommodated if, at the time the LPI header is put in service, the RCS pressure is less than 295 psig.

Answer C Discussion

Incorrect. First part is plausible since it would be correct if RCS temperature were > 250 degrees where cooldown rate is less restrictive. Second part is plausible since TS specifies LPI cooler Outlet temperature as RCS temperature when on LPI (with NO RCP's operating). Since the delta temp between cooler outlet and Tcold is in excess of the allowed cooldown rate for this RCS temperature it would be plausible to believe that the RV stress analysis limits would be exceeded since that analysis is in part driven by limiting heatup and cooldown rates. It is plausible to believe that even if cooldown rates are not violated that during the "brief period of stabilization" you could be outside the bounds of the RV stress analysis because that philosophy of allowing brief periods outside of a TS limit is consistent with other situations. As an example, the RCS is allowed to violate the RCS pressure Safety Limit for 5 minutes. That brief allowance would make it plausible to believe you would be allowed to briefly violate RV stress analysis.

Answer D Discussion

Incorrect. First part is plausible since it would be correct if RCS temperature were > 250 degrees where cooldown rate is less restrictive. Second part is correct

Basis for meeting the KA

This question is specific to the RHR system at ONS and requires information from the basis of the RCS P/T limit LCO to determine that the LPI cooler outlet temperature during the period of stabilization does not need to be considered when an LPI train is placed in service.

Basis for Hi Cog

Basis for SRO only

This question requires both application of Tech Specs and knowledge of information contained in TS basis. 10 CFR 55.43(b)(2)
It cannot be answered solely by1hr or less memory items.
It cannot be answered solely by above the line knowledge
It cannot be answered solely by knowing TS Safety Limits
It does require knowledge of TS basis that is not systems knowledge in that knowledge of how LPI cooler outlet temperatures are applied to cooldown rates
It also required application of TS's by requiring an analysis of various temperature indications and knowledge of how to apply those indication to cooldown limits provided by TS 3.4.3.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	BANK	ILT44 Q87

Development References

ILT44 Q87
ADM-TSS obj R5
TS 3.4.3 basis

Student References Provided

SYS005 2.2.25 - Residual Heat Removal System (RHRS)
SYS005 GENERIC
Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. (CFR: 41.5 / 41.7 / 43.2)

401-9 Comments:	Remarks/Status
	No miss on ILT44 exam

SYS012 A2.05 - Reactor Protection System (RPS)

Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.5)

Faulty or erratic operation of detectors and function generators ....

Given the following Unit 1 conditions:
Initial conditions:

- Reactor power = 100%
- I&E performing Reactor Protective System (RPS) calibration procedure

Current conditions:

- The RCS High Pressure trip setpoint is determined to be 6 psig non-conservative in 1A and 1B RPS Channels..

1) The actual RPS trip setpoint for RCS High Pressure is __(1)__ psig.

2) In accordance with the bases of Tech Spec 3.3.1 (Reactor Protective System (RPS) Instrumentation), the 1A and 1B RCS High Pressure Trip Functions are __(2)__.

Which ONE of the following completes the statements above?

A. 1. 2345
 2. operable

B. 1. 2345
 2. inoperable

C. 1. 2355
 2. operable

D. 1. 2355
 2. inoperable

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87

A

General Discussion

Answer A Discussion

Correct. The actual RPS trip setpoint for RCS High Pressure is 2345 psig. The allowable value per TS 3.3.1 is 2355 psig. According to TS Bases, when the trip setpoint is found to be incorrect, the trip functions are operable if the trip setpoint is within the allowable value.

Answer B Discussion

Incorrect. First part is correct. Second part is incorrect but plausible to believe the trip function is inoperable when the setpoint is found to be incorrect in the non-conservative direction and if the setpoint was 10 psig non-conservative, it would be correct.

Answer C Discussion

Incorrect. First part is incorrect. Plausible since 2355 psig is the allowable Tech Spec value. Second part is correct.

Answer D Discussion

Incorrect. First part is incorrect. Plausible since 2355 psig is the allowable Tech Spec value. Second part is incorrect but plausible to believe the trip function is inoperable when the setpoint is found to be incorrect in the non-conservative direction and if the setpoint was 10 psig non-conservative, it would be correct.

Basis for meeting the KA

Requires the ability to predict the impact of several malfunctions on RPS and to use Tech Specs to mitigate the consequences of the failures.

Basis for Hi Cog

Basis for SRO only

This question requires application of Tech Specs.
It cannot be answered solely by 1hr or less memory items.
It cannot be answered solely by above the line knowledge
It cannot be answered solely by knowing TS Safety Limits
It does require application of generic LCO requirements. 10 CFR 55.43(b)(2)

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	MODIFIED	ILT46 Q87

Development References

ILT46 Q87
IC-RPS Obj. R30, 31, 32
TS 3.3.1
TS 3.3.1 Bases

Student References Provided

SYS012 A2.05 - Reactor Protection System (RPS)

Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.5)

Faulty or erratic operation of detectors and function generators ....

401-9 Comments:

Remarks/Status

SYS013 A2.06 - Engineered Safety Features Actuation System (ESFAS)

Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations; (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Inadvertent ESFAS actuation ....

Given the following Unit 1 conditions:

- Reactor power = 100%
- RCS pressure 2155 psig stable
- RB pressure 0.1 psig stable
- ES Channel 2 actuation

1) AP/42 Enclosure 5.3 (SSF Restoration) __(1)__ required to be performed.

2) In accordance with AD-LS-ALL-0006 (Notification/Reportability Evaluation), a 4 hour ENS notification __(2)__ required.

Which ONE of the following completes the statements above?

REFERENCE PROVIDED

A. 1. is
   2. is

B. 1. is NOT
   2. is

C. 1. is
   2. is NOT

D. 1. is NOT
   2. is NOT

General Discussion

Answer A Discussion

Incorrect. First part is incorrect. Plausible since it would be correct for an ES Channel 1 actuation. Second part is incorrect. Plausible since it would be correct for a valid ES actuation.

Answer B Discussion

Incorrect. First part is correct. Second part is incorrect. Plausible since it would be correct for a valid ES actuation.

Answer C Discussion

Incorrect. First part is incorrect. Plausible since it would be correct for an ES Channel 1 actuation. Second part is correct.

Answer D Discussion

CORRECT. AP/42 Encl. 5.3 is NOT required to be performed for an ES Channel 2 actuation. It is performed following actuation of ES Channel 1. A 4 hour ENS notification is NOT required in this case, but would be required if the actuation were valid.

Basis for meeting the KA

Question requires the candidate to be able to determine, following an inadvertent ES Channel 2 actuation, if the SSF Restoration enclosure of AP/42 is required to be performed. Also requires knowledge of the Notification/Reportability Evaluation procedure and whether or not the ES Channel 2 actuation is reportable.

Basis for Hi Cog

Basis for SRO only

Per ES-401 Attachment 2 (Clarification Guidance for SRO-only Questions), requires knowledge of the NRC reporting requirements under Conditions and limitations in the facility license (10CFR55.43(b)(1).

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	NEW	

Development References

EAP-APG Obj. R9
ADM-SD Obj. R22
AP/42
AD-LS-ALL-0006

Student References Provided

AD-LS-ALL-0006

SYS013 A2.06 - Engineered Safety Features Actuation System (ESFAS)

Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations; (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Inadvertent ESFAS actuation ....

401-9 Comments:

Remarks/Status

SYS039 A2.03 - Main and Reheat Steam System (MRSS)

Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Indications and alarms for main steam and area radiation monitors (during SGTR) ....

Given the following Unit 1 conditions:

Time = 0800:

- Reactor power = 100%
- 1RIA-59 reading 10 gpm slowly increasing
- AP/31 (Primary to Secondary Leakage) is initiated

Time = 0810:

- AP/29 (Rapid Unit Shutdown) is in progress
- Reactor power = 60% decreasing
- 1RIA-59 reading 28 gpm

Time = 0845:

- Reactor is tripped
- 1A1 RCP restarted per Encl. 5.6 (RCP Restart)
- SCM = 0ºF

Time = 0847

- SCM = 0ºF

In accordance with the SGTR Tab…

1) The MINIMUM allowed power level to utilize 1RIA-59 for leakage calculation is __(1)__ percent.

2) The EARLIEST time that a transfer to the LOSCM tab is <u>required</u> is __(2)__.

Which ONE of the following completes the statements above?

A. 1. > 40
2. 0845

B. 1. > 40
2. 0847

C. 1. > 25
2. 0845

D. 1. > 25
2. 0847

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General Discussion

Answer A Discussion

Incorrect. 1st part is correct. If Reactor power is </= 40%, the EOP will not use RIA-59/60 for leakage calculation.
2nd part is incorrect. If SCM = 0 for 2 minutes after RCP start, transition to the LOSCM tab is required. If SCM = 0 for < 2 minutes, the transition to LOSCM is not required and the CRS will remain in the SGTR tab.

Answer B Discussion

CORRECT.
1st part is correct. If Reactor power is </= 40%, the EOP will not use RIA-59/60 for leakage calculation.
2nd part is correct. If SCM = 0 for 2 minutes after RCP start, transition to the LOSCM tab is required.

Answer C Discussion

Incorrect. 1st part is incorrect. Plausible because 25 is the leakage amount (in gpm) that requires entry into the EOP SGTR tab.
2nd part is incorrect. The SGTR tab will not direct transition to the LOSCM tab unless SCM = 0 for 2 minutes.

Answer D Discussion

Incorrect. 1st part is incorrect. Plausible because 25 is the leakage amount (in gpm) that requires entry into the EOP SGTR tab.
2nd part is correct. If SCM = 0 for 2 minutes after RCP start, transition to the LOSCM tab is required.

Basis for meeting the KA

Requires knowledge of how Main Steam Radiation Monitors affect procedure application during a SGTR.

Basis for Hi Cog

Basis for SRO only

ES-401, II.E
Assessment of facility conditions and selection of appropriate procedures during
normal, abnormal, and emergency situations. 10 CFR 55.43(b)(5)
Received permission from A. Goldau to use the note information to determine procedure selection on when to transfer to LOSCM tab.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References
SGTR Tab AP/31 EAP SGTR Obj R11

Student References Provided

SYS039 A2.03 - Main and Reheat Steam System (MRSS)
Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)
Indications and alarms for main steam and area radiation monitors (during SGTR) ....

401-9 Comments:

Remarks/Status
Preview

ILT48 ONS SRO NRC Examination QUESTION 90 90 **D**

SYS062 2.2.4 - AC Electrical Distribution System

SYS062 GENERIC

(multi-unit license) Ability to explain the variations in control board/control room layouts, systems, instrumentation, and procedural actions between units at a facility. (CFR: 41.6 / 41.7 / 41.10 / 45.1 / 45.13)

Given the following plant conditions:
Initial conditions:

- Unit 1 and 2 in MODE 5
- Unit 3 is at 100%

Current conditions:

- ALL units EOP Blackout tab in progress

1) Actuating Unit 1 & 2 Keowee Emergency Start Channel B switch in the Unit 2 Control Room will start __(1)__ KHU(s).

2) The switch in part 1 above __(2)__ be used to satisfy the Unit 2 Manual Keowee Emergency Start requirements of Tech Spec 3.3.22 (EPSL Manual Keowee Emergency Start Function).

Which ONE of the following completes the statements above?

A. 1. ONLY one
   2. can

B. 1. BOTH
   2. can

C. 1. ONLY one
   2. can NOT

D. 1. BOTH
   2. can NOT

General Discussion

Answer A Discussion

Incorrect. First part is incorrect. Plausible since all other switches associated with starting Keowee units in auto or manual are unit specific. Second part is plausible since it is a switch that is located in the Unit 2 Control Room and Unit 1 also has a switch on its control board

Answer B Discussion

Incorrect. First part is correct. . Second part is plausible since it is a switch that is located in the Unit 2 Control Room and Unit 1 also has a switch on its control board

Answer C Discussion

Incorrect. First part is incorrect. Plausible since all other switches associated with starting Keowee units in auto or manual are unit specific. Second part is correct.

Answer D Discussion

Correct. Any of the KHU Emergency Start switches starts both KHU's. The switch in the Unit 1 CR and the switch in the Unit 2 CR both use Unit 1's Emergency start circuits therefore the switch located in the Unit 2 Control Room can only be used to satisfy the Unit 1 TS requirements.

Basis for meeting the KA

Question requires the candidate to be able to explain the variations in control board/control room layouts pertaining to the KHU Emergency Start circuits.

Basis for Hi Cog

Basis for SRO only

Question requires knowledge of the basis of TS 3.3.22 which states that the Unit 2 switch cannot be used to satisfy the Unit 2 TS requirements.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References

EAP-BO Obj. R6
Unit 2 Encl. 5.38
Unit 3 Encl. 5.38
TS 3.3.22 basis

Student References Provided

SYS062 2.2.4 - AC Electrical Distribution System

SYS062 GENERIC

(multi-unit license) Ability to explain the variations in control board/control room layouts, systems, instrumentation, and procedural actions between units at a facility. (CFR: 41.6 / 41.7 / 41.10 / 45.1 / 45.13)

401-9 Comments:

Remarks/Status

SYS002 2.2.22 - Reactor Coolant System (RCS)
SYS002 GENERIC
Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2)

Given the following Unit 3 plant conditions:
Time = 1200

- Reactor in MODE 1
- RBNS level 20 inches increasing
- LPSW leak into RB
- Begin pumping RBNS using 3A and 3B RBNS Pumps

Time = 1205

- RBNS level as indicated below

Time = 1230

- RBNS level indication unchanged from Time = 1205

1) Condition A of Tech Spec 3.4.15 __(1)__ required to be entered.

2) __(2)__ can be used to satisfy the RIA requirements of TS 3.4.15 LCO.

Which ONE of the following completes the statements above?

REFERENCE PROVIDED

A. 1. is
2. 3RIA-47

B. 1. is
2. 3RIA-49

C. 1. is NOT
2. 3RIA-47

D. 1. is NOT

General Discussion

Answer A Discussion

Correct. TS 3.4.15 basis requires that:
Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE. With the RBNS level off scale and no way to bring it back on scale (due to the LPSW leak size) the candidate must determine that the RBNS level indication cannot perform its safety function as is therefore inoperable in the context of TS 3.4.15.
Since the LCO specifies that a particulate monitor must be used, 3RIA-47 is required. TS Bases also states RIA-47.

Answer B Discussion

Incorrect: First part is correct. Second part is plausible since it is one of the RB RIA's, however it is not a particulate monitor.

Answer C Discussion

Incorrect. First part is plausible since there is no malfunction or out of service equipment associated with the RBNS, however it cannot perform its safety function. Second part is correct.

Answer D Discussion

Incorrect. First part is plausible since there is no malfunction or out of service equipment associated with the RBNS, however it cannot perform its safety function. Second part is plausible since it is one of the RB RIA's, however it is not a particulate monitor.

Basis for meeting the KA

Requires applying Tech Spec 3.4.15 in that the SRO must assess plant conditions and apply them to the LCO requirments and the required safety function of the equipment required by the LCO to determine if the LCO is met.

Basis for Hi Cog

Basis for SRO only

Requires knowledge from the TS Bases of TS 3.4.15 to determine that the RBNS level indication cannot perform the safety function associated with identifying RCS leakage inside containment.
It cannot be answered solely by1hr or less memory items.
It cannot be answered solely by above the line knowledge
It cannot be answered solely by knowing TS Safety Limits
It does require knowledge of TS basis that is not systems knowledge

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	BANK	ILT44 Q91

Development References
ILT44 Q91 Admin-TSS obj R5 TS 3.4.15 basis

Student References Provided
TS 3.4.15

SYS002 2.2.22 - Reactor Coolant System (RCS)
SYS002 GENERIC
Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2)

401-9 Comments:

Remarks/Status
Preview - Q changed out since preview. 10/01/15. Nobody missed on ILT44.

SYS014 A2.02 - Rod Position Indication System (RPIS)
Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those 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 power to the RPIS ....

Given the following Unit 1 conditions:

- Reactor power = 100%
- Electrical malfunction results in loss of ALL Control Rod position indications.

1) In accordance with Tech Spec 3.1.7 (Position Indicator Channels) , the MAXIMUM time allowed to declare all Control Rods inoperable is __(1)__.

2) Assuming indications are NOT restored, Tech Spec 3.1.4 (Control Rod Group alignment limits), __(2)__ require reducing Reactor Power to a MAXIMUM of 60% RTP within 2 hours of declaring all Control Rods inoperable.

Which ONE of the following completes the statements above?

REFERENCE PROVIDED

A. 1. immediately
2. does

B. 1. immediately
2. does NOT

C. 1. one hour
2. does

D. 1. one hour
2. does NOT

General Discussion

Answer A Discussion

Correct. TS 3.1.7 required declaring all rods that have no position indication operable inoperable immediately. This puts you in TS 3.1.4 where both Condition A and Condition C apply. While Condition C requires Mode 3 within 12 hours, the Condition A requirement to be below 60% RTP within 2 hours would also still be applicable.

Answer B Discussion

Incorrect. First part is correct.
Second part is plausible since that requirement stems from Condition A. Condition A applies with one Control Rod inoperable. It is a common misconception when applying Tech specs that you are only in the specific CONDITION that matches plant status. i.e. I would not be in Condition A since it is for one inoperable rod because I have more than one inoperable rod. That misconception would lead the candidate to believe that only the requirements of Condition C apply. Further plausibility comes from the fact that Oconee does have an outlier SLC (16.8.3) that does work as the above "misconception" states. There is a condition for "a single battery" being inoperable and the condition only applies when there is one and ONLY one battery inoperable.

Answer C Discussion

Incorrect. First part is plausible since there are many Tech Specs that allow 1 hr as a competition time and the 1 hr CT would mean that this would still be a required memory item. Second part is correct.

Answer D Discussion

incorrect. First part is plausible since there are many Tech Specs that allow 1 hr as a competition time and the 1 hr CT would mean that this would still be a required memory item.
Second part is plausible since that requirement stems from Condition A. Condition A applies with one Control Rod inoperable. It is a common misconception when applying Tech specs that you are only in the specific CONDITION that matches plant status. i.e. I would not be in Condition A since it is for one inoperable rod because I have more than one inoperable rod. That misconception would lead the candidate to believe that only the requirements of Condition C apply. Further plausibility comes from the fact that Oconee does have an outlier SLC (16.8.3) that does work as the above "misconception" states. There is a condition for "a single battery" being inoperable and the condition only applies when there is one and ONLY one battery inoperable.

Basis for meeting the KA

Requires knowledge of procedures to mitigate consequences of loss of power to RPIS. Predicting the impact is understanding that all Control Rods must be delcared inoperable and procedure use is met by applying the requirments of TS 3.1.4.

Basis for Hi Cog

Basis for SRO only

This question requires application of Tech Specs.
It cannot be answered solely by1hr or less memory items.
It cannot be answered solely by above the line knowledge
It cannot be answered solely by knowing TS Safety Limits
It does require application of generic LCO requirements. 10 CFR 55.43(b)(2)

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References

Admin-ITS OBJ r3
TS 3.1.4
TS 3.1.7

Student References Provided

TS 3.1.4

SYS014 A2.02 - Rod Position Indication System (RPIS)
Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those 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 power to the RPIS ....

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 93 93 **D**

SYS033 2.4.9 - Spent Fuel Pool Cooling System (SFPCS)
SYS033 GENERIC
Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)

Given the following Unit 1 conditions:
Time = 1000

- Mode 6
- LPI aligned in Normal Decay Heat Removal mode
- 1A and 1B SGs in wet layup
- Fuel Transfer Canal flooded
- 1B LPI pump OOS

Time = 1030

- ALL available LPI pumps tripped and can NOT be restarted

1) At 1000, in accordance with OP/1/A/1104/004 (Low Pressure Injection System), the __(1)__ LPI pump should be operating.

2) In accordance with AP/26 (Loss of Decay Heat Removal), Enclosure __(2)__ will be initiated for heat removal.

Which ONE of the following completes the statements above?

A. 1. 1A
2. 5.18 (SSF Operation for Loss of DHR Events)

B. 1. 1C
2. 5.7 (DHR Using SF Cooling)

C. 1. 1C
2. 5.18 (SSF Operation for Loss of DHR Events)

D. 1. 1A
2. 5.7 (DHR Using SF Cooling)

General Discussion

Answer A Discussion

Incorrect. The first part is correct. The A or B LPI Pump is operated in Decay Heat Removal (DHR), if possible, per the LPI procedure since these pumps automatically restart when power is restored following a loss of power. Second part is incorrect. Plausible since this enclosure would be utilized if the Fuel Transfer Canal were not full.

Answer B Discussion

Incorrect. First part is incorrect. Plausible since it would be correct if the 1A LPI Pump were unavailable. Second part is correct. With all LPI Pumps unavailable and the Transfer Canal flooded, AP/26 will initiate Encl. 5.7 (DHR Using SF Cooling).

Answer C Discussion

Incorrect. First part is incorrect. Plausible since it would be correct if the 1A LPI Pump were unavailable. Second part is incorrect. Plausible since this enclosure would be utilized if the Fuel Transfer Canal were not full..

Answer D Discussion

Correct. With the 1B LPI Pump OOS, the A LPI Pump is operated in Decay Heat Removal (DHR), if possible, per the LPI procedure since it would automatically restart when power was restored following a loss of power. With all LPI Pumps unavailable and the Transfer Canal flooded, AP/26 will initiate Encl. 5.7 (DHR Using SF Cooling).

Basis for meeting the KA

Requires knowledge of low power/shutdown loss of Decay Heat Removal mitigation strategies as related to Spent Fuel Pool Cooling.

Basis for Hi Cog

Basis for SRO only

This question is SRO only per ES-401 Attachment 2 Clarification Guidance for SRO-only Questions due to the following:
It cannot be answered solely by using systems knowledge.
It cannot be answered solely by knowing immediate operator actions.
It cannot be answered solely by knowing entry conditions for Aps or direct entry to the EOP.
It cannot be answered solely by knowing the purpose or overall mitigation strategy of a procedure.
The question requires assessing plant conditions and then selecting the proper section of the procedure with which to proceed. 10 CFR 55.43(b)(5)

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References

EAP-APG Obj. R9
OP/1/A/1104/004 LPI
AP/26

Student References Provided

SYS033 2.4.9 - Spent Fuel Pool Cooling System (SFPCS)
SYS033 GENERIC
Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)

401-9 Comments:

Remarks/Status

GEN2.1 2.1.39 - GENERIC - Conduct of Operations
Conduct of Operations
Knowledge of conservative decision making practices. (CFR: 41.10 / 43.5 / 45.12)

Given the following Unit 1 conditions:

- 1A SG isolated due to a steam line break
- 1B SG has a tube rupture

1) Without additional station management approval, the section of the EOP that would be used to cooldown to LPI would be the section using the __(1)__SG.

2) Once RCS temperature and pressure allow alignment of LPI, the __(2)__ mode will be the INITIAL LPI alignment utilized.

Which ONE of the following completes the statements above?

A. 1. 1A
   2. normal DHR

B. 1. 1A
   2. High Pressure

C. 1. 1B
   2. normal DHR

D. 1. 1B
   2. High Pressure

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D

General Discussion

Answer A Discussion

First part is incorrect because the SGTR tab will direct you to use the SG with the tube rupture (1B) unless you have station management approval (step 61) to use the SG with the steam leak. It is plausible since it is the SG that does not have a tube leak which on the surface would logically be the SG that should cause less off site dose.

Second part is incorrect because High Pressure mode is the desired lineup unless Station management deems otherwise (Note before Step 237 SGTR tab). It is plausible because if it were Unit 3, it would be correct.

Answer B Discussion

First part is incorrect because the SGTR tab will direct you to use the SG with the tube rupture (1B) unless you have station management approval (step 61) to use the SG with the steam leak. It is plausible since it is the SG that does not have a tube leak which on the surface would logically be the SG that should cause less off site dose.

Second part is correct. Per the SGTR tab, "The credited LPI alignment for SGTR design basis event is High Pressure mode and is desired unless Station management or equipment issues deem otherwise".

Answer C Discussion

First part is correct. The SGTR tab directs using the SG with the tube leak unless station management directs using the Faulted SG.

Second part is incorrect because High Pressure mode is the desired lineup unless Station management deems otherwise (Note before Step 237 SGTR tab). It is plausible because if it were Unit 3, it would be correct.

Answer D Discussion

First part is correct. The SGTR tab directs using the SG with the tube leak unless station management directs using the Faulted SG.

Second part is correct. Per the SGTR tab, "The credited LPI alignment for SGTR design basis event is High Pressure mode and is desired unless Station management or equipment issues deem otherwise".

Basis for meeting the KA

Requires knowledge of conservative decision making as it relates to ensuring off site dose is minimized during a cooldown with both a steam leak and a SGTR in opposing SG's.

Basis for Hi Cog

Basis for SRO only

Required knowledge of approval levels required for Radiation hazards that arise during abnormal situations. This is SRO only criteria since it would re Process for release approvals. 10 CFR 55.43(b)(4)

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References

EAP-SGTR Obj R27
SGTR Tab

Student References Provided

GEN2.1 2.1.39 - GENERIC - Conduct of Operations
Conduct of Operations
Knowledge of conservative decision making practices. (CFR: 41.10 / 43.5 / 45.12)

401-9 Comments:

Remarks/Status

GEN2.1 2.1.5 - GENERIC - Conduct of Operations
Conduct of Operations
Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc. (CFR: 41.10 / 43.5 / 45.12)

In accordance with the bases of SLC 16.13.1 (Minimum Station Staffing Requirements)...

1) the NRC Communicator __(1)__ required to be a licensed or previously licensed SRO.

2) the NRC Communicator __(2)__ also be the Off-site Communicator.

Which ONE of the following completes the statements above?

A. 1. is
2. can

B. 1. is
2. can NOT

C. 1. is NOT
2. can

D. 1. is NOT
2. can NOT

General Discussion

Answer A Discussion

Correct.
Per the bases of SLC 16.13.1, an SRO serves as the offsite communicator and the NRC communicator. This is permissible since the offsite communicator role is completed prior to the NRC communicator role starting. The initial offsite communication is required within 15 minutes of the declaration and the NRC communication is required within 60 minutes.

Answer B Discussion

Incorrect.
First part is correct. Second part is incorrect. Plausible since both positions are required to be staffed and have separate duties and procedures to perform during an event.

Answer C Discussion

Incorrect.
First part is incorrect. Plausible since some ERO positions are allowed to be staffed by a licensed or previously licensed RO (TSC Operations Superintendent Assistant, Assistant to the OSC Operations Liason, and the OSM Liason).
Second part is correct.

Answer D Discussion

Incorrect.
First part is incorrect. Plausible since some ERO positions are allowed to be staffed by a licensed or previously licensed RO (TSC Operations Superintendent Assistant, Assistant to the OSC Operations Liason, and the OSM Liason).
Second part is incorrect. Plausible since both positions are required to be staffed and have separate duties and procedures to perform during an event.

Basis for meeting the KA

Requires ability to ensure the NRC Communicator and Off-site Communicator positions are properly staffed per SLC 16.31.1 (Minimum Station Staffing Requirements) as described in the bases.

Basis for Hi Cog

Basis for SRO only

In accordance with Rev. 1 of "Clarification Guidance for SRO-only Questions":
This question requires knowledge of the facility operating limitations in the TS (SLC) and their bases to determine required shift staffing. 10 CFR 55.43(b)(2)

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References

ADMIN-OMP Obj R5
OMP 2-1
OMP 1-07
ADM-TSS R5
SLC 16.13.1

Student References Provided

GEN2.1 2.1.5 - GENERIC - Conduct of Operations
Conduct of Operations
Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc. (CFR: 41.10 / 43.5 / 45.12)

401-9 Comments:

Remarks/Status

ILT48 ONS SRO NRC Examination QUESTION 96 96 **B**

GEN2.2 2.2.43 - GENERIC - Equipment Control
Equipment Control
Knowledge of the process used to track inoperable alarms. (CFR: 41.10 / 43.5 / 45.13)

In accordance with OP/0/A/1108/001 (Curves and General Information) Encl. 4.17 (Evaluation for Removal of Statalarms/Control Room Indications), which ONE of the following is the MINIMUM level of approval required to remove a Statalarm from service located on 1VB1?

A. Reactor Operator

B. Control Room Supervisor

C. Work Control Center SRO

D. Shift Manager

ILT48 ONS SRO NRC Examination QUESTION 96

96

B

General Discussion

Answer A Discussion

Incorrect. Plausible since it would be correct if asking about preparing the removal vs approving the removal. Also, specifying 1VB1 as the panel adds plausibility since it would not be one of the "front board" alarms and is therefore perceived as less critical.

Answer B Discussion

Correct. The Control Room Supervisor must approve removing a statalarm from service on any Control Room panel.

Answer C Discussion

Incorrect. Plausible since the WCC SRO, when filled by a licensed SRO, approves equipment tagouts on the unit.

Answer D Discussion

Incorrect. Plausible since there are other things in the day to day operation that require a minimum of the Shift Mangers approval.

Basis for meeting the KA

Requires generic knowledge of the process required to remove a statalarm from service.

Basis for Hi Cog

Basis for SRO only

This question meets SRO requirements since it requires knowledge of administrative processes for disabling annunciators per ES-401 Attachment 2, 10 CFR 55.43(b)(3).

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	NEW	

Development References

Admin-OMP Obj R5
1108/01

Student References Provided

GEN2.2 2.2.43 - GENERIC - Equipment Control
Equipment Control
Knowledge of the process used to track inoperable alarms. (CFR: 41.10 / 43.5 / 45.13)

401-9 Comments:

Remarks/Status

Preview

GEN2.3 2.3.15 - GENERIC - Radiation Control

Radiation Control

Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.12 / 43.4 / 45.9)

Given the following Unit 1 conditions:

Initial conditions:

- Reactor in MODE 6
- Fuel offload in progress
- Penetration openings exist due to containment penetration work in progress

Current conditions:

- Control Room notified that a Fuel Assembly has been dropped
- 1SA-8 B-9 (PROCESS MONITOR RADIATION HIGH) actuates
- 1RIA-3 (Fuel Transfer Canal Area Monitor) HIGH alarm actuates
- 1RIA-49 (RB Gas) HIGH alarm actuates

Which ONE of the following states the:

1) MAXIMUM time (minutes) allowed to isolate all containment penetrations in accordance OP/1/A/1502/009 (Containment Closure Control)?

2) Abnormal Procedure that contains the actions that are required to mitigate the initiating event?

A. 1. 30
   2. AP/1/A/1700/009 (Spent Fuel Damage)

B. 1. 30
   2. AP/1/A/1700/018 (Abnormal Release of Radioactivity)

C. 1. 35
   2. AP/1/A/1700/009 (Spent Fuel Damage)

D. 1. 35
   2. AP/1/A/1700/018 (Abnormal Release of Radioactivity)

General Discussion

Answer A Discussion

Correct.
AP/9 has direction to notify Containment Closure Coordinator to ensure any open penetrations are isolated in accordance with the Containment Closure Control procedure. These penetration must be isolated within 30 minutes therefore AP/9 would be the AP needing immediate attention.

Answer B Discussion

Incorrect:
First part is correct.
Second part is incorrect. Plausible since the entry conditions for the AP are met.

Answer C Discussion

Incorrect:
First part is incorrect. Plausible since there are other time critical actions that have to be performed in this time such as open ADVs on both Main Steam Lines within 35 minutes of a tornado.
Second part is correct.

Answer D Discussion

Incorrect:
First part is incorrect. Plausible since there are other time critical actions that have to be performed in this time such as open ADVs on both Main Steam Lines within 35 minutes of a tornado.
Second part is incorrect. Plausible since the entry conditions for the AP are met.

Basis for meeting the KA

Question requires the knowledge of Radiation alarms and automatic actions performed by each. Meets KA at SRO level since it requires the ability to prioritize radiation alarms to determine what procedure must be used.

Basis for Hi Cog

Basis for SRO only

In accordance with Clarification Guidance for SRO-only Questions, this question
requires assessing plant conditions and selecting a procedure with which to proceed to
mitigate the event. Since the entry conditions for both of the AP choices provided are met, the SRO must use knowledge of the content of the AP's to prioritize which AP should be performed first. 10 CFR 55.43(b)(5)

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	MODIFIED	ILT40 Q90

Development References

ILT40 Q90
EAP-APG obj R9
AP/9 & AP/18
OP/1502/09

Student References Provided

GEN2.3 2.3.15 - GENERIC - Radiation Control

Radiation Control

Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.12 / 43.4 / 45.9)

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:
Time = 0400

- Reactor power = 100%
- 1A GWD tank release in progress at 1/3 Station Limit

Time = 0415

- 3B GWD tank release initiated at 1/3 Station Limit

Time = 0430

- Loss of power to RM-80 skid of 1RIA-37 (Waste Gas Effluent Monitor)

1) In accordance with OP/3/A/1104/018 (GWD System) the release at 0415 is required to be approved by the __(1)__.

2) At 0430 and in accordance with the ARGs, an RO will __(2)__.

Which ONE of the following completes the statements above?

A. 1. Unit 3 CRS
2. manually close GWD-4

B. 1. Unit 3 CRS
2. verify GWD-4 has automatically closed

C. 1. SM only
2. manually close GWD-4

D. 1. SM only
2. verify GWD-4 has automatically closed

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D

General Discussion

Answer A Discussion

Incorrect. First part is incorrect. Plausible because it would be correct if the release was at 1/3 limit and only one release was in progress. Second part is incorrect. Plausible since it would be correct for loss of power to the 1RIA-45 RM-80 skid.

Answer B Discussion

Incorrect. First part is incorrect. Plausible because it would be correct if the release was at 1/3 limit and only one release was in progress. Second part is correct.

Answer C Discussion

Incorrect:. First part is correct. 2 Gaseous Waste Releases (GWRs) at 1/3 Station Limit requires Shift Manager (SM) approval. Second part is incorrect. Plausible since it would be correct for loss of power to the 1RIA-45 RM-80 skid.

Answer D Discussion

Correct:
2 Gaseous Waste Releases (GWRs) at 1/3 Station Limit requires Shift Manager (SM) approval.
Loss of power to the RIA (RM-80) skid causes any associated high alarm interlocks to actuate. In this case a loss of power to the RM-80 skid for 1RIA-37 causes GWD-4 to close to isolate the release.

Basis for meeting the KA

Question requires knowledge of the process for releasing a GWD gas tank including who has to approve the release.Understanding the process is paramount in the ability to control the release.

Basis for Hi Cog

Basis for SRO only

Per Clarification Guidance for SRO-only Questions:

Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Process for gaseous/liquid release approvals, i.e., release permits. 10 CFR 55.43(b)(4)

Additionally this question is supported as SRO by an SRO ONLY objective in the WE-GWD lesson plan.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	MODIFIED	ILT42 Q97

Development References

ILT42 Q97
OP/3/A/1104/18
ARG
WE-GWD Obj 07
RAD-RIA

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

New K/A

ILT48 ONS SRO NRC Examination QUESTION 99 99 **A**

GEN2.4 2.4.18 - GENERIC - Emergency Procedures / Plan
Emergency Procedures / Plan
Knowledge of the specific bases for EOPs. (CFR: 41.10 / 43.1 / 45.13)

Given the following Unit 1 conditions:
Time = 0000

- Reactor power = 75%
- All three seals on the1B1 RCP fail

Time = 0015

- SCM = 0ºF stable

Time = 0019

- 1A1 RCP remains operating
- LOSCM tab in progress
- Rapid depressurization of BOTH steam generators is in progress

1) The reason the 1A1 RCP is NOT secured at Time = 0019 is to __(1)__.

2) The MINIMUM condition(s) that will allow the section of the LOSCM tab to be performed that reinstates the requirement to adhere to Tech Spec cooldown rate limits is once __(2)__ is/are greater than 0 degrees F.

Which ONE of the following completes the statements above?

A. 1. prevent core uncovery that could result from phase separation if the RCP were secured
2. ALL subcooling margins

B. 1. provide forced circulation which enhances heat removal even with a two phase mixture
2. ALL subcooling margins

C. 1. prevent core uncovery that could result from phase separation if the RCP were secured
2. ANY subcooling margin

D. 1. provide forced circulation which enhances heat removal even with a two phase mixture
2. ANY subcooling margin

ILT48 ONS SRO NRC Examination QUESTION 99

99

A

General Discussion

Answer A Discussion

Correct. If a RCP remains operating greater than 2 minutes after LOSCM occurs, then it is left on. Void fraction could be greater than 70% and if the pump is subsequently stopped,
phase separation will occur and the core could be uncovered. LOSCM tab will require holding at a WHEN step (39) until all SCM's are >0 before continuing.

Answer B Discussion

Incorrect. First part is incorrect. Plasuible because it would be correct if SCM was lost for < 2 minutes. Second part is correct

Answer C Discussion

Incorrect. First part is correct. Second part is plausible since there are various conditions in the EOP which take actions basesd soley on Core SCM (Ex. Transfer to ICC tab) and actions are taken based on any subcooling margin such as securing RCPs if any SCM is 0.

Answer D Discussion

Incorrect. First part is incorrect. Plasuible because it would be correct if SCM was lost for < 2 minutes.Second part is plausible since there are various conditions in the EOP which take actions basesd soley on Core SCM (Ex. Transfer to ICC tab) and actions are taken based on any subcooling margin such as securing RCPs if any SCM is 0.

Basis for meeting the KA

This question requires knowing the basis for specfic steps in the EOP.

Basis for Hi Cog

Basis for SRO only

This question requires assessing plant conditions using that assessment to determine if or when a section of the LOSCM tab can be performed.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	BANK	ILT45 Q77

Development References

ILT45 Q77
EAP-LOSCM Obj. 15
EOP Rule 2

Student References Provided

GEN2.4 2.4.18 - GENERIC - Emergency Procedures / Plan
Emergency Procedures / Plan
Knowledge of the specific bases for EOPs. (CFR: 41.10 / 43.1 / 45.13)

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) In accordance with RP/0/A/1000/002 (Control Room Emergency Coordinator Procedure) implementation of the OSAG __(1)__ require the use of 10CFR50.54(x) and(y) provisions.

2) If implemented, 10CFR50.54(x) will require the MINIMUM level of approval to be __(2)__ .

Which ONE of the following completes the statements above?

A. 1. does
2. a licensed SRO

B. 1. does
2. the Emergency Coordinator

C. 1. does NOT
2. a licensed SRO

D. 1. does NOT
2. the Emergency Coordinator

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General Discussion

Answer A Discussion

Correct. The first part of the question is correct in that any implementation of the OSAG requires a 10CFR50.54(x) declaration.
The second part of the question is correct in that the MINIMUM level of approval for a 10CFR50.54(x) situation, per 10CFR50.54(y) is a licensed SRO.

Answer B Discussion

Incorrect distractor. First part of distractor 'B' is correct, second part of distractor 'B' is incorrect. Second part of distractor 'B' is plausible because the Emergency Coordinator is the highest level authority on site during an emergency, is granted the authority to make a 50.54(x) declaration, and probably would be the decision-maker in this situation. However, technically speaking the second part of this distractor is incorrect because the Emergency Coordinator is not the MINIMUM required level of approval; instead, 10CFR50.54(y) specifies that the minimum level of authorization is a licensed SRO.

Answer C Discussion

Incorrect distractor. First part of distractor 'C' is incorrect, second part of distractor 'C' is correct. First part of distractor 'C' is plausible because no other directed procedural transition requires a 50.54(x) declaration. A SRO applicant may incorrectly believe that use of approved procedures (OSAG) is not a departure from the license basis condition, and therefore may choose this incorrect distractor.

Answer D Discussion

Incorrect distractor. Both parts of this distractor are incorrect. For plausibility descriptions, see above discussions of distractor 'B' and 'C.'

Basis for meeting the KA

This question matches the K/A in that, given an extreme emergency situation (similar to Fukushima Dai-ichi), the question requires the SRO applicant to demonstrate knowledge of the requirements associated with OSAG entry and implementation of 10CFR50.54(x) and (y). These decisions are associated with the Emergency Coordinator function in the Emergency Plan implementation procedures.

Basis for Hi Cog

Basis for SRO only

This question is linked to 10CFR55.43(b)(1), Conditions and limitations in the facility license, in that SRO-level authorization, at a minimum, is required to transition to the OSAG and declare a 10CFR50.54(x) condition. Licensed ROs are not allowed to make that determination.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	BANK	ILT43 Q100

Development References

ILT43 Q100
EAP-SEP Obj. R10
RP/0/A/1000/002

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