

Question 1

Unit 1 plant conditions are as follows:

- Unit 1 is operating at 100% power with both recirc pumps operating at 94% speed
- OPRM inoperable
- PICSY is unavailable
- 'B' recirculation pump RPT breaker 1A20601 trips, placing the plant in single loop operations
- ON-164-001, Loss of Reactor Recirculation Flow has been entered and operator actions are in progress
- A maintenance worker calls the control room to report bumping into the 1A20601 RPT breaker and seeing the breaker trip

Current plant conditions:

- Reactor power: 58%
- 'A' recirculation pump speed: 71%
- Core pressure drop as indicated on XR-14301: 6.5%

Based upon the given plant condition, select the appropriate course of action:

- A. Restart the 'B' Recirculation Pump
- B. Place the Reactor Mode Switch in SHUTDOWN
- C. Immediately exit the region of the Power/Flow map IAW ON-178-002
- D. Raise the speed of the 'A' Recirculation Pump >85%

Answer:

A is incorrect. Cannot exit region by restarting the tripped recirculation pump.

B is correct. Power/Flow map requires immediate SCRAM if in Region I and OPRM inoperable.

C is incorrect. If candidate either makes a calculational/plotting error for core flow, or does not note that OPRM are inoperable, will select to immediately exit the region.

D is incorrect. TS 3.4.1 and ON-178-002 Caution (1) requires the operating loop recirculation pump speed $\leq 80\%$.

K/A Rating: 295001 AK1.02 3.3/3.5

K/A Statement: Knowledge of the operational implications of power/flow distribution as it applies to Partial or Complete Loss of Forced Core Flow Circulation

References: TM-OP-064C, Reactor Recirc
ON-164-002, Rev. 33
Unit 1 COLR
ON-178-002, Rev. 16
TS 3.4.1 Rev. 3
GO-100-009 Attachment B Rev. 19

Applicant Ref:
Power/Flow Map with
stability region info
blanked out
GO-100-009 Att. B

Learning Objective: TM-OP-064C: 2593e/f, 2546, 2558m

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR 41.8 to 41.10

Comments:

QUESTION 2

Unit 1 & 2 are at 100%. Aux buses 11A and 11B are energized from SU bus 10 via startup transformer 10.

The plant experiences a loss of Montour Mountain tie line.

Assuming all automatic actions occur as designed, and no other operator actions occur, what is supplying power to Aux buses 11A and 11B?

- A. Startup Bus 20 via Startup Transformer 20 (OX104) to Bus 11A and 11B via breakers 10204, 10104, and tie breaker 10502.
- B. Unit Aux Transformer 11 to Bus 11A and 11B via breakers 10201 and 10101.
- C. Unit Aux Transformer 12 to Bus 11A and 11B via breakers 10204, 10104, and tie breaker 10502.
- D. Both Aux buses are de-energized, tie breaker 10502 must be manually closed.

K&A Rating: 295003AK301 (3.3/3.5)

K&A Statement: **Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : Manual and auto bus transfer**

Justification:

- A. **Incorrect but plausible:** If the applicant does not recall the tie breaker 10502 configuration, and no auto bus transfer occurs for aux buses when Montour Mountain tie line is lost.
- B. **Incorrect but plausible:** If the applicant does not recall that no auto bus transfer occurs for aux buses when Montour Mountain tie line is lost.
- C. **Incorrect but plausible:** If the applicant does not recall that no auto bus transfer occurs for aux buses when Montour Mountain tie line is lost and the tie breaker does not auto close to bring power from Unit 2 UAT.
- D. **Correct;** Both aux buses will remain de-energized, no auto bus transfer occurs. Tie breaker must be manually closed.

References: TM-OP-003-ST, 13.8 kV
Distribution System, Rev. 05

Student Ref: NONE

Learning Objective: 10054

Question source: Modified from SQ bank
AD045/1357/ 2

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.5 to 45.6

Comments:

Question 3

With Station 125 VDC in a normal line-up, a system fault results in the loss of 1D634, 125 VDC Bus, which is now de-energized. Following the DC power loss, an Emergency Diesel Generator LOCA start signal is received.

Which of the following describes the impact of this loss of DC power on the 'D' Diesel Generator?

- A. Will automatically start, but can **NOT** be manually started
- B. Will automatically start, and can be manually started
- C. Will **NOT** automatically start, but can be manually started from the Engine Control Panel 0C521, 'C' Start pushbutton in Local Mode
- D. Will **NOT** automatically start, but can be manually started from the Control Room Panel 0C653, 'C' Start pushbutton in Droop Mode

Answer:

A is incorrect. There is no effect on the 'D' Diesel Generator for a loss of 1D634, normal DC power for 'D' Diesel Generator is 1D644.

B is correct. There is no effect on the 'D' Diesel Generator for a loss of 1D634, normal DC power for 'D' Diesel Generator is 1D644.

C is incorrect. There is no effect on the 'D' Diesel Generator for a loss of 1D634, normal DC power for 'D' Diesel Generator is 1D644. The 'C' Diesel Generator will not automatically start and cannot be manually started (remotely or locally), since all 'C' DG logic power is from 1D634 and is de-energized. Alternate power must be aligned to the 'C' Diesel Generator.

D is incorrect. There is no effect on the 'D' Diesel Generator for a loss of 1D634, normal DC power for 'D' Diesel Generator is 1D644. The 'C' Diesel Generator will not automatically start and cannot be manually started (remotely or locally), since all 'C' DG logic power is from 1D634 and is de-energized. Alternate power must be aligned to the 'C' Diesel Generator.

K/A Rating: 295004 AA2.04 3.2

K/A Statement: Ability to determine and/or interpret system lineups as they apply to partial or complete loss of DC power.

References: ON-102-630, Rev. 10
TM-OP-024-ST, Rev. 10 EDG A-D
TM-OP-024-FS, Rev. 3

Applicant Ref: None

Learning Objective: TM-OP-024: 2072b, 2260g

Question source: Modified from Susquehanna
bank #1411

Question History: Modified from 5/08 SQ Exam
and bank

Cognitive level: Memory/Fundamental knowledge: X
Comprehension/Analysis:

10CFR55 41.10/43.5/45.13

Comments:

QUESTION 4

Given the following:

- Reactor power is 25%
- A Main Turbine trip occurs
- Three bypass valves fail to open

Which of the following describes the initial response of Reactor pressure and Reactor level?

Initially, reactor pressure will ___(1)___ and Reactor level will ___(2)___.

- A. (1) rise
(2) rise
- B. (1) rise
(2) lower
- C. (1) lower
(2) rise
- D. (1) lower
(2) lower

K&A Rating: 295005AK1.03 (3.5/3.7)

K&A Statement: **Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR TRIP: Pressure effects on reactor level**

Justification:

- A. **Incorrect but plausible** part (1) is correct but part (2) is incorrect. Plausible if candidate believes the reactor will scram and/or does not understand the fluid dynamics.
- B. **Correct:** At 25% power, the reactor will NOT scram on turbine trip. With only two BPVs available, reactor power will exceed two BPV capability (~10.4%) resulting in initial pressure rise which will compress voids and cause level to lower.
- C. **Incorrect:** Part (1) and (2) are incorrect. Plausible if candidate believes the reactor will scram and/or does not understand the fluid dynamics.
- D. **Incorrect but plausible:** Part (1) is incorrect, part (2) is correct. Plausible if candidate believes the reactor will scram and/or does not understand the fluid dynamics.

References: TM-OP-083-ST, Main Steam System, Student Ref: NONE
Rev. 10 Page 28

Learning Objective: TM-OP-083-OB, 1651

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.8 to 41.10

Comments:

Question 5

Unit 1 was initially operating at 60% power. A loss of vacuum occurred and the reactor was shutdown in accordance with ON-100-101 SCRAM, SCRAM IMMINENT. Due to errors in level control, reactor water level reached -57", resulting in SBLC injection for level control.

Plant conditions are as follows:

- The reactor is shutdown with power at 6E-02% and lowering slowly
- Multiple control rods failed to insert
- Reactor water level is +24" and steady
- Drywell pressure is 1.27# and rising slowly

Per EO-100-102 "RPV Control" and EO-100-113 "Level/Power Control", under these conditions RPV depressurization is _____.

- A. Allowed and re-criticality may occur
- B. Allowed and re-criticality will NOT occur
- C. NOT allowed until all control rods have been inserted
- D. NOT allowed until Cold Shutdown Boron Weight has been injected

Answer:

A is correct. Per EO-100-113 step LQ/P-8, cooldown is permitted under these conditions because the reactor is subcritical on rod position, although subsequent re-criticality may occur due to the positive reactivity added during cooldown.

B is incorrect. Per EO-100-113 step LQ/P-8, cooldown is permitted under these conditions because the reactor is subcritical on rod position, although subsequent re-criticality may occur due to the positive reactivity added during cooldown.

C is incorrect. Per EO-100-113 step LQ/P-8, cooldown is permitted under these conditions because the reactor is subcritical on rod position.

D is incorrect. Per EO-100-113 step LQ/P-8, cooldown is permitted under these conditions because the reactor is subcritical on rod position.

K/A: 295006 G2.4.9 3.8/4.2

K/A Statement: Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

References: EO-100-113 Sheet 1 Rev. 10
EO-100-113 Bases Rev. 8

Applicant Ref: None

Learning Objective: PP002: 14613

Question source: Modified Peach Bottom
12/2009 Question 55

Question History: Similar to 1/11 SQ exam
question 79

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.10/43.5/45.13

Comments:

Modified

QUESTION 6

Unit 1 control room has been evacuated due to a fire with the following:

- The immediate actions of ON-100-009, Control Room Evacuation, have been taken
- Control of Unit 1 has been established at the Remote Shutdown Panel
- RPV pressure is at 690 psig
- RPV level is at +32 inches

Which one of the following describes the response of the condensate system?

- A. The condensate pumps will inject into the RPV
- B. The condensate pumps will NOT inject into the RPV until RPV level reaches +30 inches
- C. The condensate pumps will need to be manually started from the remote shutdown panel since auto initiation signal is disabled when control of Unit 1 was established from the remote shutdown panel
- D. The condensate pumps will NOT inject into the RPV until RPV pressure reaches 650 psig.

K&A Rating: 295016AA1.06 (4.0/4.1)

K&A Statement: **Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT** : Reactor water level.

Justification:

- A. **Correct:** If RPV pressure drops below 700 psig, condensate pumps will inject when RPV level < +35 inches due to condensate pumps not installed on the remote shutdown panel.
- B. **Incorrect but plausible:** Plausible if the applicant does not recall the RPV level setpoint.
- C. **Incorrect but plausible:** Plausible if the applicant believes that the condensate pumps are protected by the remote shutdown panel and does not understand or demonstrate knowledge of the components on the remote shutdown panel..
- D. **Incorrect but plausible:** Plausible if the applicant does not recall the RPV pressure setpoint.

References: ON-100-109

Student Ref: NONE

Learning Objective: TM-OP-044-OB, Rev. 0, 10723

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.7 to 45.6

Comments:

Question 7

Unit 1 is operating at 68% power with the following conditions:

'B' RBCCW pump is out of service for bearing replacement.

A small leak develops on the suction of the in-service 'A' RBCCW pump.

Annunciator AR-123-E03, RBCCW PUMPS DISCHARGE HEADER LO PRESS, is in alarm.

RBCCW supply header temperature is 95°F and rising slowly.

ON-114-001 Loss of RBCCW, has been entered

In accordance with ON-114-001, which one of the following components should be removed from service first to reduce the heat load on the RBCCW system?

- A. RB Sump Coolers
- B. CIG Compressors
- C. Reactor Recirc Pumps
- D. RWCU System

Answer:

A is incorrect. RB sump coolers are not a significant enough heat load on the system to warrant isolation

B is incorrect. CIG compressors are not a significant enough heat load on the system to warrant isolation

C is incorrect. Reactor recirc pump seal coolers, bearing oil coolers and thermal cover coolers are significant loads on the system and are monitored IAW ON-114-001. Actions are taken to reduce speed or take them out of service based upon meeting thresholds, but are not directed to be immediately removed from service whereas the RWCU system is.

D is correct. The RWCU system NRHX is the largest heat load on the system. By removing RWCU from service, this allows more time to troubleshoot/repair the cooling problem before further action is required

K/A Rating: 295018 AK1.01 3.5/3.6

K/A Statement: Knowledge of the operational implications of the effects on component/system operations as it applies to Partial or Complete Loss of Component Cooling Water

References: ON-114-001, Rev. 22

Applicant Ref: None

Learning Objective: TM-OP-014-OB 1676i

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehension/Analysis:

10CFR55 41.8 to 41.10

Comments:

QUESTION 8

With Unit 1 at full power, Instrument Air Header pressure is slowly degrading.

In accordance with the Immediate Operator Actions of ON-118-001, Loss of Instrument Air.

As Instrument air header pressure lowers:

- 1) At what instrument air pressure is the RO required to SCRAM the reactor?
- 2) State the basis for this action?

- A.
 - 1) ≤ 75 psig
 - 2) Since MSIVs will begin to drift closed, causing a pressure transient and scram.
- B.
 - 1) ≤ 75 psig
 - 2) Since Scram Inlet and Outlet Valves will drift open, causing control rods to insert.
- C.
 - 1) ≤ 65 psig
 - 2) Since MSIVs will begin to drift closed, causing a pressure transient and scram.
- D.
 - 1) ≤ 65 psig
 - 2) Since Scram Inlet and Outlet Valves will drift open, causing control rods to insert.

K&A Rating: 2.4.11 (4.0/4.2)

K&A Statement: Knowledge of abnormal condition procedures.

Justification:

- A. **Incorrect but plausible:** Reactor scram is not required until IA pressure reaches 65 psig. MSIV closure is not the reason, since this is an analyzed transient and scram. This is plausible, since MSIVs will drift closed (< 50 psig) and cause a significant transient.
- B. **Incorrect but plausible:** Plausible if the applicant does not recall the IA pressure setpoint. If IA pressure reduces scram inlet and outlet valves will drift open causing control rods to insert.
- C. **Incorrect but plausible:** MSIV closure is not the reason, since this is an analyzed transient and scram. This is plausible, since MSIVs will drift closed (< 50 psig) and cause a significant transient.
- D. **Correct:** Reactor scram is not required until IA pressure reaches 65 psig. If IA pressure reduces scram inlet and outlet valves will drift open causing control rods to insert.

References: ON-118-001, Rev. 24, Pages 3 and 21, Applicant Ref: NONE
TM-OP-018-ST, Page 4

Learning Objective: TM-OP-018-OB, Rev. 0, 1768

Question source: SQ Bank

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.10/43.5/45.13

Comments:

Question 9

Unit 1 has just entered mode 5 with water level <22' above the flange in the beginning hours of a refueling outage.

An unidentified Shutdown Cooling Outboard Isolation signal has occurred, resulting in a loss of Shutdown Cooling. ON-149-001, "Loss of RHR Shutdown Cooling Mode" has been entered. It has been determined that reactor coolant temperature cannot be maintained $\leq 200^{\circ}\text{F}$ and the Reactor Cavity cannot currently be flooded.

Plant conditions have been established as follows:

- RPV Pressure is 0 psig
- RPV water level is 38"
- SRV's C and F are open
- Reactor coolant temperature is 159°F
- Current reactor coolant heatup rate is $29^{\circ}\text{F}/\text{hour}$
- 'A' loop of RHR is being placed in Suppression Pool cooling mode

Which of the following describes the action required to establish a means of decay heat removal in accordance with ON-149-001?

Establish RPV injection using _____(1)_____ in order to maintain RPV water level _____(2)_____.

- (1) One core spray pump with suction from the CST
(2) $>45''$
- (1) Two core spray pumps with suction from the CST
(2) $>45''$
- (1) One core spray pump with suction from the Suppression Pool
(2) $130''-180''$
- (1) Two core spray pumps with suction from the Suppression Pool
(2) $130''-180''$

Answer:

A is incorrect. Two core spray pumps with suction from suppression pool are required per procedure. 45” is a plausible distractor as this is where water level is raised to enhance natural circulation

B is incorrect. Two core spray pumps with suction from suppression pool are required per procedure. 45” is a plausible distractor as this is where water level is raised to enhance natural circulation

C is incorrect. Two core spray pumps with suction from suppression pool are required per procedure. Level band is correct, and allows flooding of main steam lines to allow flow through SRV's to return to suppression pool.

D is correct. Two core spray pumps with suction from suppression pool are required per procedure. Level band is correct, and allows flooding of main steam lines to allow flow through SRV's to return to suppression pool.

K/A: 295021 AA1.04 Loss of Shutdown Cooling 3.7/3.7

K/A Statement: Ability to operate and or monitor Alternate Heat Removal Methods as they apply to loss of Shutdown Cooling

References: ON-149-001, Rev. 23

Applicant Ref: None

Learning Objective: TM-OP-049-OB Rev. 2: 192bb
AD045: 15304/15307/15308

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.7/45.6

Comments:

QUESTION 10

Unit 1 in a refueling outage with the following:

- A fuel bundle, being moved from its core location to the Spent Fuel Pool, has just been dropped from the grapple.
- The dropped bundle is now in the transfer canal.
- REFUEL FLOOR HI EXH HI-HI RADIATION annunciator (E03) is alarming.

Which one of the following identifies the actions required by ON-081-001, Fuel Handling Accident?

- A. Evaluate conditions for EO-100-104, Secondary Containment Control Entry.
- B. Direct security to monitor reactor building card readers, and ensure all personnel in the reactor building are evacuated.
- C. Notify the Shift Manager to initiate a protected area evacuation.
- D. Verify the SGBT system starts.

K&A Rating: 295023 AK2.03 (3.4/3.6)

K&A Statement: **Knowledge of the interrelations between REFUELING ACCIDENTS and the following:** Radiation monitoring equipment.

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not recall that Hi-Hi radiation AR-106-001 operator actions direct evaluation of conditions for EO-100-104 entry.
- B. **Incorrect but plausible:** Plausible if the applicant does not recall that ON-081-001 directs security to inactivate RB ingress readers if the radiological conditions warrants.
- C. **Incorrect but plausible:** Plausible if the applicant does not realize that refuel floor and reactor building, if necessary, are to be evacuated vice a protected area evacuation.
- D. **Correct:** Per ON-070-001, one of the automatic action which is needed to be verified is start of the SGBT system.

References: ON-081-001, Rev. 10
AR-106-001, Rev. 47

Applicant Ref: NONE

Learning Objective: 1991, TM-OP-070-OB, LO Systems, Standby Gas Treatment System Rev. 02

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X

10CFR 41.7/45.8

Comments:

Question 11

Following a transient on Unit 1, the following conditions exist:

- | | |
|--------------------------------------|---------------------|
| - Drywell pressure | 7.9 psig and rising |
| - Drywell temperature | 140 °F and rising |
| - Suppression Chamber pressure | 7.8 psig and rising |
| - Suppression Pool water temperature | 82 °F and steady |

Which of the following describes what has occurred?

- A. A Safety Relief Valve Tailpipe has broken in the Suppression Chamber air space with the respective SRV lifted and open.
- B. A reactor recirculation partial line break has occurred.
- C. A Safety Relief Valve has opened and stuck open.
- D. A steam line break in the Drywell has occurred with a pair of Suppression Chamber to Drywell Vacuum Breakers open.

Answer:

A is incorrect. This would be energy deposition into the suppression chamber without a suppression function and would be indicated by suppression chamber pressure leading drywell pressure

B is incorrect. Drywell pressure would be leading suppression chamber pressure significantly

C is incorrect. Suppression pool temperature would be rising. Drywell and suppression chamber pressures are not consistent with this condition.

D is correct. Drywell and suppression chamber are approximately equal, with drywell pressure leading slightly

K/A Rating: 295024 EA2.04 (3.9/3.9)

K/A Statement: Ability to determine and/or interpret Suppression chamber pressure as it applies to high drywell pressure

References: EO-100-103 Bases Rev. 8

Applicant Ref: None

Learning Objective: TM-OP-059-OB Rev. 1: 10360h

Question source: Modified SSES ILO Bank
#2009

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.10/43.5/45.13

Comments:

QUESTION 12

Unit 1 is operating at 100% power when steam pressure Channel 'A' fails upscale.

Which one of the following describes the plant response to this malfunction?

- A. The Reactor scrams on high RPV pressure.
- B. The Reactor scrams on a MSIV isolation.
- C. The Reactor remains at power. The steam pressure Channel 'B' controls pressure at a new, lower value.
- D. The Reactor remains at power. The steam pressure Channel 'B' controls pressure at a new, higher value.

Question 13

Unit 2 has just experienced a transient. Plant conditions are as follows:

- ATWS
- NSSS MSIV Isolation
- Reactor power is 34%
- Suppression Pool level is 21.5 feet
- Suppression Pool cooling is not available

Which one of the following limits is challenged by these conditions?

- A. Pressure Suppression Pressure
- B. Heat Capacity Temperature Limit
- C. RHR & CS Vortex Limit
- D. Primary Containment Pressure Limit.

Answer:

A is incorrect. PSP is not a concern since there are no given conditions that indicate the Primary Containment is not functioning correctly.

B is correct. The given conditions indicate SRV discharge into the Suppression Pool with no cooling available. This will challenge HCTL and ultimately Primary Containment integrity.

C is incorrect. While Suppression Pool level is lower than normal, RHR & CS pump maximum flows to prevent vortexing are not impacted until below a Suppression Pool level of 21 feet.

D is incorrect. PCP is not a concern because there is no given condition of Primary Containment high pressure.

K/A: 295026 EK3.02 3.9/4.0

K/A Statement: Knowledge of the reasons for Suppression Pool Cooling as it applies to Suppression Pool high water temperature

References: EO-100-103 Bases Rev. 8

Applicant Ref: None

Learning Objective: TM-OP-059-OB Rev. 1: 10360k

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehension/Analysis:

10CFR55 41.5/45.6

Comments:

QUESTION 14

A seismic event has occurred and the following conditions exist on Unit 1:

- Loss of Offsite Power (LOOP) occurred.
- Steam leak occurred inside the Drywell.
- RPV Level is -91 inches; lowering 2 inches/minute.
- RPV Pressure is 720 psig; lowering 10 psig/minute.
- DW Temperature is 285°F; rising at 5°F/minute.
- Drywell Pressure is 18.2 psig; rising.
- Suppression Pool Temperature is 104°F; rising.
- Suppression Chamber Pressure is 13.1 psig; rising.

Which one of the following EOP actions is required?

- A. Initiate suppression spray followed by drywell spray.
- B. Initiate RHR in drywell spray Mode ONLY.
- C. Rapid Depressurization for LPCI injection.
- D. Depressurize using SRVs and maintain level between +13" and +54" using condensate system.

K&A Rating: 295028 2.4.6 (3.7/4.7)

K&A Statement: **Knowledge of the EOP mitigation strategies.**

Justification:

- A. **Correct:** Given the above conditions, EO-100-103, Primary Containment Control, PC/P-7 and/or DW/T-5 requires spraying the Drywell to lower containment pressure required to control Primary Containment parameters. (when suppression pressure > 13 psig and before avg DW temp reaches 340°F). But also, before spraying the DW, Supp Chamber will be required to be spray when pressure is above 13 psig.
- B. **Incorrect but plausible:** Plausible if the applicant does not recall that suppression pressure is above 13 psig and suppression chamber spray is required prior to DW spray.
- C. **Incorrect but plausible:** Plausible if the applicant does not recall the LPCI injection pressure, and also parameters that require rapid depressurization.
- D. **Incorrect but plausible:** Plausible if the applicant does not recognize the initial condition of LOOP, which takes away condensate capability.

References: EO-000-103, Rev. 7, Pages 27 and 45, Applicant Ref: NONE
EO-000-102, Rev. 9, Page 19,

Learning Objective: 14586, EOP learning objective

Question source: Susquehanna bank.

Question History: None

Cognitive level: Memory/Fundamental knowledge
Comprehension/Analysis: X

10CFR 41.10/43.5/45.13

Comments:

Question 15

Due to concrete degradation, Unit 1 Suppression Chamber wall develops an unisolable leak, resulting in Suppression Pool level dropping to and stabilizing at the leak location at 12' 2".

Which one of the following is correct?

- A. Downcomer flow can be quenched.
- B. RCIC turbine exhaust can be condensed.
- C. SRV steam flow will NOT be completely condensed.
- D. Suppression pool water temperature CANNOT be determined.

Answer:

A is correct. Downcomers opening uncover at 12 feet.

B is incorrect. RCIC exhaust begins to uncover at approximately 17 feet.

C is incorrect. SRV tailpipes begin to uncover at five feet.

D is incorrect. The four lower detectors are located at three feet.

K/A Rating: 295030 EK2.07 (3.5/3.8)

K/A Statement: Knowledge of the interrelations between Low Suppression Pool water level and Downcomer submergence

Reference: TM-OP-059
EO-100-102 Bases Pages 13-18

References: EO-100-103 Bases Rev. 8

Applicant Ref: None

Learning Objective: TM-OP-059-OB Rev. 1: 10356d

Question source: Modified SSES ILO Bank
#2002

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehension/Analysis:

10CFR55 41.7/45.8

Comments:

Question 16

A plant transient has resulted in the following:

- Reactor pressure is 150 psig and steady
- 3 SRVs are open
- RPV water level is -200" and steady
- Core Spray loop A is injecting at 6420 gpm

Which one of the following describes the status of core cooling?

- A. Partial core submergence and spray flow with NO assurance of adequate core cooling
- B. Partial core submergence and spray flow ensures adequate core cooling
- C. Steam Cooling with injection ensures adequate core cooling
- D. Steam Cooling with injection DOES NOT ensure adequate core cooling

K&A Rating: 295031 EA2.04 (4.6/4.8)

K&A Statement: **Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL:** Adequate core cooling.

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not recall the adequate core cooling criteria for partial submergence. Flow requirements (≥ 6350 gpm) with water level restored and maintained at or above the jet pump suctions (-210”).
- B. **Correct:** A loop of core spray injecting per design flow requirements (≥ 6350 gpm) with water level restored and maintained at or above the jet pump suctions (-210”) indicates partial core submergence and spray flow which is adequate to keep the core cool.
- C. **Incorrect but plausible:** Plausible if the applicant does not recall the minimum steam cooling pressure with 3 SRVs open is 319 psig. For Steam cooling with injection the RPV pressure must be maintained at or above the minimum steam cooling pressure.
- D. **Incorrect but plausible:** Plausible if the applicant does not recall the minimum steam cooling pressure with 3 SRVs open is 319 psig. For Steam cooling with injection the RPV pressure must be maintained at or above the minimum steam cooling pressure.

References: EO-200-114
EO-000-102, Rev. 9 Basis

Applicant Ref: NONE

Learning Objective: 14591 Determine if adequate core cooling exist. (EOP learning objective)

Question source: NMP 2009 Exam modified

Question History: None

Cognitive level: Memory/Fundamental knowledge
Comprehension/Analysis: X

10CFR 41.10/43.5/45.13

Comments:

Question 17

The following conditions exist:

- An ATWS has occurred with power at 30 percent
- ARI and scram have been reset
- The SDV is partially drained

Which one of the following methods will result in the largest differential pressure across the CRD mechanism operating piston for inserting a control rod?

- A. Drifting control rods with cooling water pressure
- B. Single rod insertion using maximum drive pressure
- C. Single rod insertion using normal drive pressure
- D. Individual scram of control rod with SRI switch

Answer:

A is incorrect. Drifting control rods in using cooling water pressure would result in the lowest CRD piston D/P (approximately 60 psid)

B is incorrect. Maximum drive pressure would result in a relatively high CRD piston D/P (350 psid as directed by EO-100-113 step CR-7)

C is incorrect. Using normal drive pressure would result in a CRD piston D/P of approximately 250 psid.

D is correct. Individual scram of control rods results in the highest CRD piston D/P of approximately 1500 psid from charging header to depressurized SDV for a short period.

K/A Rating: 295037 EK2.05 4.0/4.1

K/A Statement: Knowledge of the interrelations between SCRAM condition present and reactor power above APRM downscale or unknown and the CRD hydraulic system

References: EO-100-113 Bases Rev. 8
TM-OP-055 Rev. 6

Applicant Ref: None

Learning Objective: TM-OP-055-OB Rev. 2: 2413

Question source: SSES ILO Bank #1055

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehension/Analysis:

10CFR55 41.7/45.8

Comments:

Question 18

A liquid radwaste discharge is in progress. The following alarm is observed in the radwaste and main control room:

- Sample Flow Low FAL 06433

Which one of the following automatic actions would occur?

- A. No automatic actions occur
- B. Effluent Rad monitor sample pump trips ONLY.
- C. Plant Effluent Inbd and Outbd isolation valves HV-643A1 & A2 close ONLY.
- D. Plant Effluent Inbd and Outbd isolation valves HV-643A1 & A2 close and Effluent Rad monitor sample pump trips.

K&A Rating: 295038 (3.7/3.9)

K&A Statement: **Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE** : Process liquid radiation monitoring system.

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not recall the automatic actions of isolation of the Outbd and Inbd Iso Valve HV-06432A1& A2 Closes. Also the effluent rad monitor sample pump trips.
- B. **Incorrect but plausible:** Plausible if the applicant does not recall the automatic actions. Also, the isolation of the Outbd and Inbd Iso Valve HV-06432A1& A2 occurs by valves closing.
- C. **Incorrect but plausible:** Plausible if the applicant does not recall the automatic actions. Also, the rad monitor sample pump trips.
- D. **Correct:** Both automatic actions occur. Outbd and Inbd Iso Valve HV-06432A1& A2 Closes, and the effluent rad monitor sample pump trips.

References: TM-OP-079E-ST, Process Radiation Monitoring, Rev. 03. Applicant Ref: NONE

Learning Objective: TM-OP-079E-OB, LO Systems, Process Radiation Monitoring, Rev 02, 1200

Question source: New.

Question History: None

Cognitive level: Memory/Fundamental knowledge: X

10CFR 41.7/45.6

Comments:

Question 19

A fire has just occurred outside the control room. The plant has been shutdown in accordance with ON-100-101 SCRAM, SCRAM Imminent due to the fire. While firefighting is in progress, the RCIC turbine initiates and cannot be overridden or isolated. RPV water level is 35" and rising. What actions are required by ON-013-001 and what is the reason for those actions?

<u>Action</u>	<u>Reason</u>
A. Close MSIV	Prevent damage to Main Steam Lines
B. Close MSIV	Prevent damage to RCIC turbine
C. Depressurize RPV	Prevent damage to RCIC turbine
D. Depressurize RPV	Prevent damage to SRV tailpipes

Answer:

A is incorrect. The MSIV are used as Appendix R safe shutdown paths

B is incorrect. The MSIV are used as Appendix R safe shutdown paths

C is correct. The RPV is depressurized prior to reactor water level reaching 118" and the main steam lines to prevent damage to the RCIC turbine

D is incorrect. A calculation has been performed that proves that the SRV tailpipes will not be damaged by water or two phase flow.

K/A Rating: 600000 AA2.17 (3.1/3.6)

K/A Statement: Ability to determine and interpret systems that may be affected by the fire as it applies to Plant Fire on Site.

Reference: ON-013-001 Revision 29

References: ON-013-001 Rev. 29

Applicant Ref: None

Learning Objective: TM-OP-050-OB Rev. 1: 2015e

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehension/Analysis:

10CFR55 None listed

Comments:

Question 20

Unit 1 is at 98% power and power is being raised IAW GO-100-002 and OP-164-002 with 'A' RRP operating in fine speed control and 'B' RRP operating in monitoring mode. The following condition is observed on the Main Generator Capability Curve.

Operating point on the curve is within 2 MWe of the generator capability curve.

Which of the following describes the automatic response of the ICS/DCS system?

'A' RRP Controller

'B' RRP Controller

- | | |
|--|--|
| A. will go on HOLD for up to 60 minutes | will decrease Rx power in ~ 4 MWth increments until sufficient margin has been established |
| B. will go on HOLD for up to 60 minutes | speed will remained unchanged |
| C. will continue to increase Rx Power in ~ 4 MWth increments every 12 minutes, until the selected overall power increase value has been achieved | speed will remained unchanged |
| D. will terminate the power increase demand request | will decrease Rx power in ~ 4 MWth increments until sufficient margin has been established |

K&A Rating: 700000 2.1.28 (4.1)

K&A Statement: **Knowledge of the purpose and function of major system components and controls.**

Justification:

- A. **Incorrect but plausible:** plausible because if the operating point on curve is within ≤ 2.1 MWE of the generator capability curve, then the power change is placed on HOLD for up to 60 minutes but for the 'B' controller to reduce power in ~ 4 MWth increments MWe Available to the Generator Capability Curve must be ≤ 0.1 MWe.
- B. **Correct:** If the operating point on curve is within ≤ 2.1 MWE of the generator capability curve, then the power change is placed on HOLD for up to 60 minutes.
- C. **Incorrect but plausible:** Plausible if the applicant does not recall that ≤ 2.1 MWE of the generator capability curve, then the power change is placed on HOLD for up to 60 minutes. If there was no HOLD place then the normal power increase would occur in this manner.
- D. **Incorrect but plausible:** Plausible if the applicant does not recall that ≤ 2.1 MWE of the generator capability curve, then the power change is placed on HOLD, it does not terminate. When monitor mode initiates a speed decrease, it also places a termination request to the other operating fine speed controller, but this occurs at ≤ 0.1 MWE.

References: TM-OP-095D-FS, Main Generator Cooling, Rev. 03. OP-164-002, Rev. 4 Applicant Ref: NONE

Learning Objective: 1115, TM-OP-095D-OB, LO Systems, Generator Hydrogen Cooling and Gas Supply Rev 0.

Question source: New.

Question History: None

Cognitive level: Memory/Fundamental knowledge X
Comprehension/Analysis:

10CFR 41.7

Comments:

Question 21

Unit 1 is conducting a reactor startup. Plant conditions are as follows:

- Reactor power is 17%
- Total feedwater flow is 16.8%
- Reactor Pressure is 940 psig
- Difficulties in level control cause a slow swing from +35" to +11"

Based upon the above conditions, at this moment, reactor recirculation pump speed is currently restricted to _____(1)_____ due to _____(2)_____.

- A. (1) 30%
(2) Total Feedwater flow
- B. (1) 30%
(2) RPV water level
- C. (1) 48%
(2) RPV water level
- D. (1) 48%
(2) Individual RFP Flow

Answer:

A is incorrect. Total feedwater flow is >16.4% but RPV water level is less than the speed limiter 1 setpoint of 13 inches.

B is correct. Current recirculation pump speed is restricted to 30% due to RPV water level reaching the speed limiter 1 setpoint of 13 inches.

C is incorrect. Although the 48% limiter is active due to individual reactor feed pump flow <16.4% for the other non-feeding reactor feed pumps, reactor recirculation pump speed is already restricted to the 30% speed limiter due to current plant conditions.

D is incorrect. The 48% limiter is active due to individual reactor feed pump flow <16.4% for the other non-feeding reactor feed pumps, reactor recirculation pump speed is already restricted to the 30% speed limiter due to current plant conditions.

Limiter 1 (30%):

- Reactor low water level (13")
- Total feedwater flow <16.4 percent for >20 seconds
- Recirc pump discharge valve not fully open
- Manual initiation

Limiter 2 (48%):

- Reactor low water level (30") coincident with HI-HI FW heater level in heaters 1 or 2
- Circ water or condensate pump trip
- Individual reactor feedwater pump flow <16.4 percent
- Manual initiation

K/A: 295009 AK 2.03 3.1/3.2

K/A Statement: Knowledge of the interrelations between Low Reactor Water Level and the recirculation system

References: TM-OP-064E Rev. 3

Applicant Ref: None

Learning Objective: TM-OP-064E-OB Rev. 0: 16021

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:

Comprehension/Analysis: X

10CFR55 41.7/45.8

Comments:

Reference PB 2/2007 Question 22

Question 22

Given the following conditions:

- The plant was at 100% power when a 60 gpm LOCA occurred
- Drywell pressure is 2.0 psig and rising 0.1 psig every two minutes
- NO operator actions have been taken

What is the status of the Drywell Floor Drain sump pumps?

- A. All four sump pumps stopped.
- B. All four sump pumps running continuously.
- C. All four sump pumps running intermittently depending on sump level.
- D. Two sump pumps running continuously and the other two sump pumps running intermittently depending on sump level.

K&A Rating: 295010 (3.6)

K&A Statement: **Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE** : Drywell floor and equipment drain sumps.

Justification:

- A. **Correct:** Drywell floor drain sump isolation valves close on high drywell pressure signal (1.79 psig).
- B. **Incorrect but plausible:** plausible if the applicant does not recall that high drywell pressure signal isolates all four sumps and interprets that the 60 gpm leak would give Hi-Hi sump levels in both sumps and it is beyond capacity of all four pumps. Capacity of one pump (30 gpm).
- C. **Incorrect but plausible:** plausible if the applicant does not recall that high drywell pressure signal isolates all four sumps and interprets that the 60 gpm leak would give Hi-Hi sump levels in both sumps and it is NOT beyond capacity of all four pumps. Capacity of one pump (30 gpm).
- D. **Incorrect but plausible:** plausible if the applicant does not recall that high drywell pressure signal isolates all four sumps and interprets that the 60 gpm leak would give Hi-Hi sump levels in both sumps and it is NOT beyond capacity of all four pumps. Capacity of one pump (30 gpm).

References: Training materials for drywell and equipment drains. Applicant Ref: NONE

Learning Objective:

Question source: New.

Question History: None

Cognitive level: Memory/Fundamental knowledge: X

10CFR 41.7/45.6

Comments:

Question 23

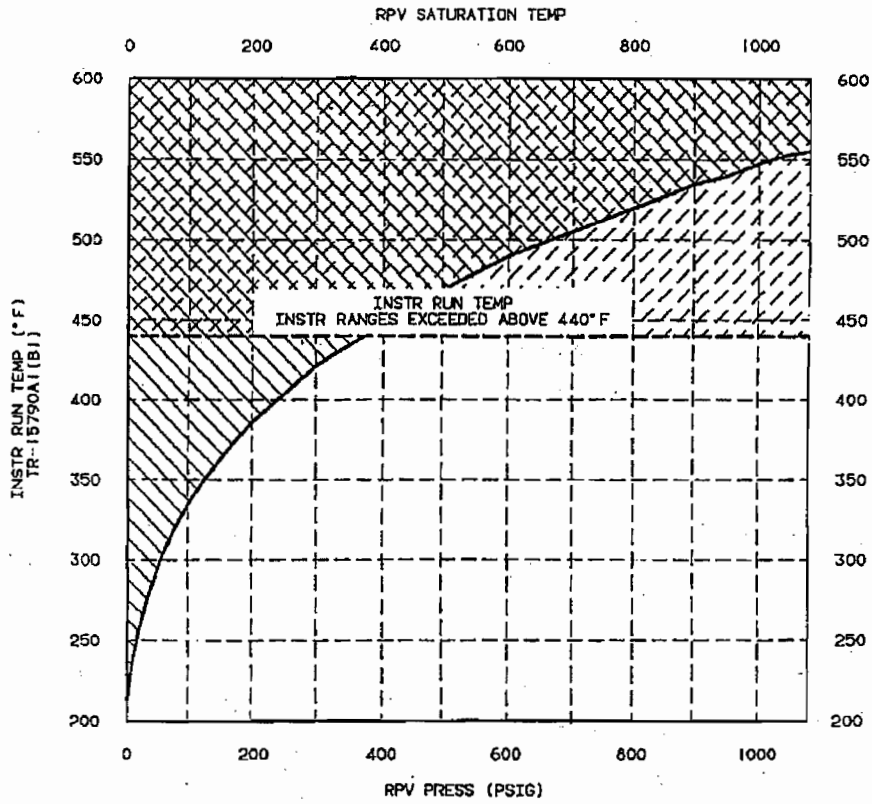
Unit 1 has experienced a steam leak in the drywell concurrent with a fire on elevation 749. Rapid depressurization in accordance with EO-100-112 is in progress due to being unable to restore and maintain drywell temperature <340°F. Plant conditions are as follows:

- All control rods are inserted
- Drywell Pressure is 30 psig
- Drywell temperature is 345°F
- All high pressure feed sources are unavailable
- Multiple failures prevented injection by LPCI and Core Spray
- RPV pressure is 95 psig and lowering
- NRLA=0" NRLB=0" NRLC=1"
- Wide Range A=-126" Wide Range B=-123"
- Fuel Zone A=-301" Fuel Zone B=-296"

Based upon the above conditions and attached tables, what is the required action?

- A. Line up and vent the drywell by using Standby Gas Treatment System
- B. Enter EO-100-114 RPV Flooding and raise level until flooded to main steam lines
- C. Bypass DW cooling isolations and operate all available DW coolers
- D. Restore RPV level >-161" using all available injection systems/subsystems

FIG 1 SAT



SHUTDOWN

UPSET

EXTENDED

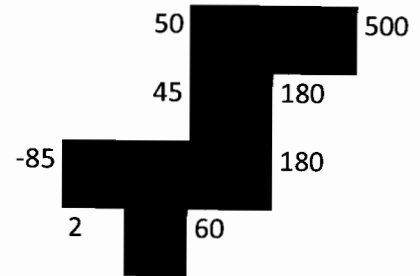
NARROW

WIDE

NO FIRE ON 749

X218_Z7(X232_Z5) ALARM

FUEL ZONE



-145 60

-125 60

-290 -110

Answer:

A is incorrect. Normal DW venting not authorized >1.72 psig

B is incorrect. RPV flooding is not required; RPV level is known because Wide Range B is above minimum indicated level

C is incorrect because bypassing DW cooler isolations is not permitted during a LOCA IAW ES-134-001

D is correct. With a known RPV level, RC/L of EO-100-102 will direct restoring RPV level using available injection systems.

K/A: 295012 AK1.01

K/A Statement: Knowledge of the operational implications of the pressure/temperature relationship is it applies to High Drywell Temperature

References: EO-100-102 Bases Rev. 8
ON-145-004 Rev. 18
ES-134-001 Rev. 17

Applicant Ref: None

Learning Objective: PP002: 14618

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:

Comprehension/Analysis: X

10CFR55 41.8 to 41.10

Comments:

Question 24

The plant is operating at 100 percent power.

Which one of the following malfunctions will result in a stable higher power level once steady state conditions are achieved?

- A. Main Condenser Circulating Water Pump trip
- B. Output of 'A' Pressure Regulator fails High
- C. Inadvertently isolating the Reactor Water Cleanup System
- D. An Extraction Steam Bleeder Trip Valves trips on high feedwater heater level

Question 25

Given the following conditions:

- Unit 1 is at 60% power for waterbox cleaning and control rod pattern adjustment
- Control rods 18-43, 18-19, 42-43, and 42-19 have just been inserted for maintenance
- 1B CRD pump is currently blocked for repair of a lube oil leak
- At 2235, 1A CRD pump trips on overcurrent
- At 2239, accumulator trouble alarms are received for HCU 30-31 and 42-43
- At 2241, charging header pressure <940 psig
- At 2242, NPO reports HCU pressure for HCU 30-31 and 42-43 are 925 psig
- At 2244, accumulator trouble alarms are received for HCU 18-15 and 22-31, also confirmed with NPO at 930psig

In accordance with ON-155-007, Loss of CRD System Flow, based upon the above conditions:

- (1) What action is required?
- (2) What is the latest time that action can be taken to remain in compliance with procedure?

- (1) Perform ON-100-101, SCRAM, SCRAM IMMINENT
(2) 2302
- (1) Perform ON-100-101, SCRAM, SCRAM IMMINENT
(2) 2304
- (1) Place the mode switch in SHUTDOWN
(2) 2302
- (1) Place the mode switch in SHUTDOWN
(2) 2304

Answer:

A is incorrect. ON-155-007 step 3.2.4 and TS 3.1.5 directs placing the mode switch in SHUTDOWN immediately once 20 minutes has expired with charging header pressure <940 psig and 2 or more accumulators inoperable. At 2302, the charging header pressure has been less than 940 psig but only one accumulator is inoperable because control rod 42-43 is already fully inserted.

B is incorrect. ON-155-007 step 3.2.4 and TS 3.1.5 directs placing the mode switch in SHUTDOWN immediately once 20 minutes has expired with charging header pressure <940 psig and 2 or more accumulators inoperable.

C is incorrect. ON-155-007 step 3.2.4 and TS 3.1.5 directs placing the mode switch in SHUTDOWN immediately once 20 minutes has expired with charging header pressure <940 psig and 2 or more accumulators inoperable. At 2302, the charging header pressure has been less than 940 psig but only one accumulator is inoperable because control rod 42-43 is already fully inserted.

D is correct. ON-155-007 step 3.2.4 and TS 3.1.5 directs placing the mode switch in SHUTDOWN immediately once 20 minutes has expired with charging header pressure <940 psig and 2 or more accumulators inoperable. At 2304, the charging header pressure has been <940 psig for 23 minutes, and the second & third accumulators are declared inoperable, requiring action to place the mode switch in SHUTDOWN.

K/A: 295022 G2.1.23 4.3/4.4

K/A Statement: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

References: TM-OP-055-FS Rev. 1
ON-155-007 Rev. 21

Applicant Ref: None

Learning Objective: TM-OP-055-OB Rev. 2: 13107

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.10/43.5/45.2/45.6

Comments:

PB 12/2008 Question 62

Question 26

Unit 1 is operating at 100% power when the following conditions occur:

- RCIC Equipment Room Area Temperature goes into alarm.
- RCIC Equipment Room Area Temperature reading is 120°F and rising.
- A steam leak in the RCIC room has been confirmed.
- EO-000-104, "Secondary Containment Control" has been entered.
- SGTS SPING release rate exceeds Hi-Hi Alarm.
- Zone 3 High Radiation Isolation signal is received
- SGTS FAN 0V109B is not running.

What is the plant response and required operator actions if RCIC room temperature continues rising and exceeds 150°F.

	Plant Response	Required Operator actions
A.	RCIC will automatically isolate	Re-establish Reactor Building HVAC
B.	RCIC will be required to be manually isolated	Start SGTS Fan 0V109B
C.	RCIC will automatically isolate	Start SGTS Fan 0V109B
D.	RCIC will be required to be manually isolated	Re-establish Reactor Building HVAC

K&A Rating: 295032EK1.02 (3.6/4.0)

K&A Statement: **A Knowledge of the operational implications of the following concepts as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE:** Radiation releases.

Justification:

- A. **Incorrect but plausible:** Plausible if applicant does not recall the automatic isolation setpoint for the RCIC isolation based on room temperature reaching 167°F. Also re-establishing RB HVAC is performed when the SGTS SPING release rates is below Hi-Hi alarm.
- B. **Correct:** RCIC will be manually isolated since an automatic isolation occurs at 167°F and since SGTS FAN 0V109B is not in the correct initiation status IAW ON-159-002 and ES-070-001 SGTS FAN 0V109B will need to be placed in the required isolation position.
- C. **Incorrect but plausible:** Plausible if applicant does not recall the automatic isolation setpoint for the RCIC isolation based on room temperature reaching 167°F. Also, SGTS FAN 0V109B is not in the correct initiation status IAW ON-159-002 and ES-070-001 SGTS FAN 0V109B will need to be placed in the required isolation position.
- D. **Incorrect but plausible:** Plausible if the applicant does not recall the re-establishing RB HVAC is performed when the SGTS SPING release rates is below Hi-Hi alarm.

References: EO-000-104, Rev. 12
ON-159-002, Rev. 29

Applicant Ref: NONE

Learning Objective:

Question source: New.

Question History: None

Cognitive level: Memory/Fundamental knowledge
Comprehension/Analysis: X

10CFR 41.8 / 41.10

Comments:

Question 27

With Unit 2 operating at 100% power, the following conditions exist:

- The watertight door between HPCI and RCIC was left open with a hose running through it for maintenance support
- AR-114-H03 HPCI Pump Room Flooded alarm is in
- AR-108-H03 RCIC Pump Room Flooded alarm is in
- NPO dispatched to the scene reports a steam/water leak upstream of HV155-005, HPCI discharge check valve and reports HPCI/RCIC room levels at 28" and rising slowly
- Attempts to close HPCI discharge line inboard and outboard isolation valves HV155-F006 and F007 have been unsuccessful

Which of the following actions are required by EOP's?

- A. Perform a plant shutdown in accordance with GO-200-004
- B. SCRAM the reactor and go to EO-200-102, RPV Control ONLY
- C. SCRAM the reactor and Rapid Depressurization in accordance with EO-200-112
- D. Open all bypass valves irrespective of cooldown rate

Answer:

A is incorrect. A primary system is discharging into the secondary containment.

B is incorrect. With a primary system discharging into the secondary containment and two areas above max safe level, a scram and rapid depressurization is required.

C is correct. With a primary system discharging into the secondary containment and two areas above max safe level, a scram and rapid depressurization is required.

D is incorrect. At the time of the report from the NPO, the room water levels are already above max safe. Opening all bypass valves irrespective of cooldown rate is performed in RPV control and done prior to reaching a rapid depressurization threshold. This allows using the condenser as a heat sink to depressurize the reactor vice the containment.

K/A: 295036 EK3.01

K/A Statement: Knowledge of the reasons for emergency depressurization as it applies to Secondary Containment High Sump/Area Water Level

References: EO-100-104 Bases Rev. 6

Applicant Ref:
EO-100-104 with no
entry conditions

Learning Objective: PP002: 14594

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.5/45.6

Comments:

Question 28

Unit 2 experienced a LOCA inside the primary containment. EO-000-112, "Rapid Depressurization" is in progress with all 6 ADS Safety Relief Valves open. Current conditions are as follows:

- RPV level -170 inches
- RPV pressure 390 psig
- Drywell pressure 10 psig
- All ECCS pumps are running

Based on the above conditions, RHR Outboard Injection valve F015A is __ (1) __ and RHR Testable Check valve F050A is __ (2) __.

- A. (1) open, (2) open.
- B. (1) closed, (2) open.
- C. (1) open, (2) closed.
- D. (1) closed, (2) closed.

K&A Rating: 203000K5.01 (2.7/2.9)

K&A Statement: **Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI:** Testable check valve operation.

Justification:

- A. **Incorrect but plausible:** Incorrect but plausible if the applicant does not recall RHR pump shutoff head pressure (204 psig).
- B. **Incorrect but plausible:** Incorrect but plausible if the applicant does not recall RHR pump shutoff head pressure (204 psig), and interlock auto opening logic of RHR outboard injection valve.
- C. **Correct:** RHR outboard injection valve auto opens at -129"/1.72 psig AND Low RPV pressure ≤ 420 psig, and the testable check valve will remain closed due to Rx pressure above RHR pump shutoff pressure (204 psig).
- D. **Incorrect but plausible:** Incorrect but plausible if the applicant does not recall RHR pump shutoff head pressure (204 psig), and interlock auto opening logic of RHR outboard injection valve.

References: TM-OP-049-ST RHR, Rev. 07

Applicant Ref: NONE

Learning Objective: TM-OP-049-OB, LO Systems, Residual Heat Removal, Rev. 02, 2066

Question source: New.

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis X

10CFR 41.5/45.3

Comments:

Question 29

Unit 1 is currently in mode 5 with the refueling cavity flooded and fuel pool gates removed. Plant conditions are as follows:

- Fuel moves are being performed in the spent fuel pool
- 1B RHR pump in shutdown cooling
- RPV water temperature is 127°F and steady
- All other low pressure ECCS in a normal alignment

A bus lockout of ESS Bus 1B occurs; which RHR pump(s) and associated flowpath(s) are capable of being lined up to provide shutdown cooling?

- A. None
- B. 1D RHR Pump
- C. 1A and 1C RHR Pump
- D. 1A, 1C, and 1D RHR Pump

Answer:

A is correct. While A, C, and D RHR pumps have power, the loss of ESS Bus 1B also causes a loss of 'B' RPS, closing RHR shutdown cooling suction valves F008 and F009, causing a complete loss of SDC from either loop.

B is incorrect. While 1D RHR pump has electrical power, the loss of 'B' RPS closes RHR shutdown cooling suction valves F008 and F008, causing a complete loss of SDC from either loop.

C is incorrect. While 1A and 1C RHR pumps have electrical power, the loss of 'B' RPS closes RHR shutdown cooling suction valves F008 and F008, causing a complete loss of SDC from either loop.

D is incorrect. Loss of ESS Bus 1B causes 1B RHR pump to trip on loss of power. Loss of ESS Bus 1B also causes a loss of 'B' RPS, closing RHR shutdown cooling suction valves F008 and F009, causing a complete loss of SDC from either loop.

K/A 205000 K6.01 3.3/3.4

K/A Statement: Knowledge of the effect that a loss or malfunction of A.C. electrical power will have on the Shutdown Cooling System (RHR Shutdown Cooling Mode)

References: ON-158-001 Rev 17
TM-OP-058-FS Rev. 4

Applicant Ref: None

Learning Objective: TM-OP-058-OB Rev. 0: 10072
TM-OP-058-OB Rev. 0: 2487
TM-OP-059-OB Rev. 1: 2142f

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.7/45.7

Comments:

See SSES Bank question 1977

Question 30

HPCI is being run for the quarterly full flow test surveillance. The reactor operator has just reached the required 5000 gpm flow rate with the controller in AUTO when the ramp generator circuit output signal fails low.

Which one of the following describes the response, if any, of the HPCI System?

- A. HPCI speed continually rises until overspeed trip actuates.
- B. HPCI speed continually lowers to idle and flow lowers to minimum.
- C. HPCI speed and flow remain the same.
- D. HPCI trips due to a loss of speed reference signal.

Question 31

Given the following conditions:

- An ATWS is in progress on Unit 1
- Reactor Power is 21%
- RPV Water Level is being maintained -60" → -110"
- AR-107-001 (A03) SBLC SQUIB VALVES LOSS OF CKT CONTINUITY alarm is in
- No squib valve position indication is available
- SBLC has not yet been initiated

Which of the following describes (1) the status of the SBLC system, and (2) what corrective actions must be taken to inject SBLC, if any?

- A. (1) ONLY SBLC Pump 2A is available
(2) Initiate SBLC per OP-153-001, "Standby Liquid Control System"
- B. (1) ONLY SBLC Pump 2B is available
(2) Initiate SBLC per OP-153-001, "Standby Liquid Control System"
- C. (1) BOTH SBLC subsystems are available
(2) Initiate SBLC per OP-153-001, "Standby Liquid Control System"
- D. (1) NEITHER SBLC subsystem is available
(2) Implement ES-150-002, "Boron Injection Via RCIC"

Answer:

A is incorrect. While the loss of position indication would also cause a loss of power to the 'B' squib firing circuit, there is no loss of power to the SBLC pumps. Each squib valve can provide 100% flow. Misconception of pump and squib valve power supplies would have student believe that a SBLC pump also lost power.

B is incorrect. While the loss of position indication would also cause a loss of power to the 'B' squib firing circuit, there is no loss of power to the SBLC pumps. Each squib valve can provide 100% flow. Misconception of pump and squib valve power supplies would have student believe that a SBLC pump also lost power.

C is correct. While the loss of position indication would also cause a loss of power to the 'B' squib firing circuit, there is no loss of power to the SBLC pumps. Each squib valve can provide 100% flow, and the system will function as designed.

D is incorrect. Both SBLC pumps have power and there is power to at least one squib valve firing circuit. System is capable of providing 100% design flow and performing its design function.

Due to the requirements of NUREG-1021, RO outline Tier 2/Group 1 (Form ES-401-1) requires selection of 26 exam question K/A's from a total pool of 23 K/A's. As such, certain system K/A's are sampled twice. In this case, 211000 SBLC is sampled in both K4 and A2. The K4 question is assessing the candidate's ability to determine what conditions would indicate SBLC injection success. The A2 question assesses the candidate's ability to determine, based upon annunciator and panel indication, whether the SBLC system has the ability to initiate and inject as designed, and the procedure required to inject SBLC into the vessel. These questions are sufficiently different, although testing the same system, test different concepts related to the system.

From TM-OP-053-FS:

"A" Pump powered from 1B236 and "B" Pump from 1B217

From AR-107-001 (A03) SBLC SQUIB VALVES LOSS OF CKT CONTINUITY:

1. PROBABLE CAUSE:

- 1.1 System initiation.
- 1.2 Blown control power fuses F1A, F2, F3, F4, F5, F6, F23, or F24 for squib valves at Panel 1C617 in Upper Relay Room.
- 1.3 Decomposition of igniter elements in either explosive valve.
- 1.4 Loss of 'A' Squib Firing Power 1Y23621.
- 1.5 Loss of 'B' Squib Firing Power 1Y21607.
- 1.6 Loss of Squib Valve Position Indication 1Y21607

K/A: 211000 A2.03 Standby Liquid Control System 3.2/3.4

K/A Statement: Ability to (a) predict the impacts of the following on the Standby Liquid Control System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. Power Failures.

References: TM-OP-053-ST Rev. 10
AR-107-001 Rev. 29

Applicant Ref: None

Learning Objective: TM-OP-053-OB Rev. 1: 1221b/10095d/10094e/1214c

Question source: Modified

Question History: Modified from 1/11 SQ Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.5/45.6

Comments:

Question 32

Given the following:

- Unit 1 is operating at 100% power
- Loss of 125 VDC to Core Spray Div II occurs
- Investigation reveals a blown fuse on 125 VDC power distribution panel 1D624001

- Subsequently, a LOCA occurs, resulting in the following plant conditions:
- RPV level is -75 inches
- RPV pressure is 410 psig
- Drywell pressure is 4.5 psig

Which one of the following describes the status of the Core Spray System?

'A' Core Sprav Loop

'B' Core Spray Loop

- | | |
|---|---|
| A. Both pumps ON; injection valve is OPEN | Both pumps ON; injection valve is OPEN |
| B. Both pumps ON; injection valve is CLOSED | Both pumps OFF; injection valve is CLOSED |
| C. Both pumps ON; injection valve is OPEN | Both pumps OFF; injection valve is CLOSED |
| D. Both pumps ON; injection valve is CLOSED | Both pumps ON; injection valve is CLOSED |

Question 33

Unit 1 has just experienced a loss of level control and reactor scram. AR-109-A04 ECCS Loop A RX LO LEVEL alarm is in. Plant conditions are as follows:

- HPCI is out of service for Aux Oil Pump maintenance
- RCIC is injecting at full flow in Automatic
- RPV pressure is 940# and steady
- RPV water level is -132" and lowering slowly
- RHR Pump A is running
- Core Spray Pump A is running
- Drywell Area Unit Coolers have tripped
- EDG A/C are running unloaded

Based upon the above conditions, what actions are required by the operator in accordance with AR-109-A04 ECCS Loop A RX LO LEVEL?

- A. Inhibit ADS
- B. Start B/D EDG's ONLY
- C. Start RHR Pump C and Core Spray Pump C ONLY
- D. Start RHR Pump C, Core Spray Pump C, and B/D EDG's

Answer:

A is incorrect. ADS function is desired since minimal high pressure injection is available and evidenced by Rx water level lowering. Applicant may choose this if they believe alarm is erroneous.

B is incorrect. Only 'A' and 'C' EDG start for this alarm. Applicant may believe that all four EDG should be running due to reactor water level being below ECCS setpoint.

C is correct. Automatic actions from this alarm are:

- RHR Loop A initiates (Pumps A/C)
- Core Spray Loop A initiates (Pumps A/C)
- Automatic Depressurization System receives a low level initiation signal
- Diesel Generators 'A' & 'C' Auto start
- Drywell Area Unit Coolers trip

D is incorrect. C RHR and C Core Spray pumps should be started. B/D EDG should not be running for the given conditions. Applicant may believe that all four EDG should be running due to reactor water level being below ECCS setpoint.

K/A 209001 2.4.50 Low Pressure Core Spray System 4.2/4.0

K/A Statement: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

References: AR-109-001 Rev. 26

Applicant Ref: None

Learning Objective: TM-OP-024-OB Rev. 1: 2260o
TM-OP-049-OB Rev. 2: 192r
TM-OP-051-OB Rev. 1: 2080g
TM-OP-083E-OB Rev. 1: 2100

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehension/Analysis:

10CFR55 41.10/43.5/45.3

Comments:

Question 34

Unit 1 scrammed from 100% power due to a loss of vacuum and is experiencing an ATWS.

Current plant conditions are as follows:

Reactor power is 36%

RPV level is +25 inches and steady

RPV pressure is 950 psig and is being controlled via SRVs.

Suppression pool temperature is 97°F and rising.

The CRS has directed the RO to inject Standby Liquid Control (SBLC). RO initiates SBLC system using the 1P208A pump.

Which of the following indications listed below indicates that SBLC system is successfully injecting into the core?

- A. SBLC Tank level is decreasing and SBLC SQUIB VLVS LOSS OF CKT CONTINUITY annunciator is extinguished
- B. SBLC Pump A discharge pressure is 1150 and SBLC SQUIB READY A White indicating light is extinguished, while SBLC SQUIB READY B White indicating light is lit.
- C. SBLC Tank level is decreasing and both SBLC SQUIB READY A-B White indicating lights are lit.
- D. SBLC Pump A discharge pressure is 1150 and both SBLC SQUIB READY A-B White indicating lights are extinguished.

K&A Rating: 211000K4.08 (4.2)

K&A Statement: **Knowledge of STANDBY LIQUID CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following:**
System initiation upon operation of SBLC control switch.

Justification:

- A. **Incorrect but plausible:** Incorrect but plausible if the applicant does not understand the loss of continuity circuit for the squib valves. The annunciator should be lit.
- B. **Incorrect but plausible:** Incorrect but plausible if the applicant believes that the SBLC squib ready white indicating light is extinguished only for the A loop.
- C. **Incorrect but plausible:** Incorrect but plausible if the applicant does not understand the design of the SBLC squib ready white indicating lights. Actually, both of the lights are extinguished.
- D. **Correct:** Pump discharge pressure increases to approx 200 psig above Rx pressure and Both Squib Ready Lights extinguish indicating both squib valves fired.

References: OP-153-001, Rev. 27
TM-OP-053FS, Rev. 02

Applicant Ref: NONE

Learning Objective:

Question source: Bank Modified.

Question History: None

Cognitive level: Memory/Fundamental knowledge: X

10CFR 41.7

Comments:

Question 35

Given the following plant conditions:

- Unit 1 is at 74% power
- The electric plant is in a normal at-power lineup
- RPS 'A' is aligned to its normal RPS MG Set power supply
- RPS 'B' is aligned to its alternate power supply
- An internal fault in S/U Trans T-10 has caused a loss of S/U Bus 10
- Breaker 1A20109 fails to automatically close

No operator actions have been taken. Which of the following describes the status of RPS one minute later?

- A. NO Scram
- B. HALF Scram on RPS 'A'
- C. HALF Scram on RPS 'B'
- D. FULL Scram

Answer:

A is incorrect. This condition causes a loss of ESS Bus 1A and loss of power to RPS 'A' longer than can be sustained by the flywheel, causing a HALF scram on RPS 'A'.

B is correct. Loss of S/U Bus 10 causes a loss of power to Essential MCC 1B217, causing a loss of power to the 'A' RPS MG set. The normal power supply breaker 1A20101 will trip on UV, but the alternate power supply breaker 1A20109 fails to close, resulting in a power loss longer than the RPS MG set can maintain output.

C is incorrect. No effect on RPS 'B'; the alternate power supplied is fed from Aux Bus 11B.

D is incorrect. No effect on RPS 'B'; this transient causes a loss of power to RPS 'A' due to failure of the alternate power supply breaker to close.

K/A: 212000 K2.01 3.2/3.3

K/A Statement: Knowledge of electrical power supplies to the RPS motor-generator sets

References: TM-OP-058-ST Rev. 9
TM-OP-004-ST Rev. 2

Applicant Ref: None

Learning Objective: TM-OP-004-OB Rev. 2: 10121/2239b
TM-OP-058-OB Rev. 0: 10072

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.7

Comments:

New

Question 36

Unit 2 is operating at five percent power with the mode switch in STARTUP. The following neutron monitoring indications exist:

- APRM CHANNEL '1' is reading five percent.
- APRM CHANNEL '2' is reading five percent.
- APRM CHANNEL '4' is reading four percent.
- IRM CHANNEL 'A' is reading 105 on Range 8.
- IRM CHANNEL 'G' is reading 115 on Range 8.
- IRM CHANNEL 'H' is reading 118 on Range 8.

WHICH of the following describes the automatic actions that should occur?

- A. Rod block, but no half-scam.
- B. Rod block and full scam.
- C. No rod block and no half-scam.
- D. No automatic actions should occur.

K&A Rating: 211000K1.01 (3.9)

K&A Statement: **Knowledge of the physical connections and/or cause-effect relationships between INTERMEDIATE RANGE MONITOR (IRM) SYSTEM and the following: RPS**

Justification:

- A. **Correct:** IRM H is below its scram setpoint (120) and IRM G and H are above the rod block setpoint (108).
- B. **Incorrect but plausible:** Incorrect but plausible if the applicant does not recall the scram setpoint. Applicant may think that a full scram is a possibility since both trains of channels are listed in the initial conditions.
- C. **Incorrect but plausible:** Incorrect but plausible if the applicant does not recall the scram and rod block setpoints.
- D. **Incorrect but plausible:** Incorrect but plausible if the applicant does not recall the scram and rod block setpoints.

References: TM-OP-078B-FS, IRM, Rev. 02 Applicant Ref: NONE

Learning Objective: 2339

Question source: Bank Modified.

Question History: None

Cognitive level: Memory/Fundamental knowledge: X

10CFR 41.2 to 41.9 / 45.7 to 45.8

Comments:

Question 37

Unit 1 is currently in a refueling outage. Plant conditions are as follows:

- Reactor is in Mode 5 with NO core alterations in progress
- Shutdown cooling is in service using 'B' RHR Pump
- Shorting links are removed
- Mode switch is in REFUEL
- Control rod 14-19 is at position 12; all others are at 00
- Fuel is being moved in the spent fuel pool
- AR-104-B06, "SRM Upscale or Inop" is received
- SRM 'D' is currently reading $3.5E5$ cps

Which of the following describes the status of RPS and control rod 14-19?

- A. No scram signals are generated and control rod 14-19 may continue to be moved
- B. No scram signals are generated and control rod 14-19 cannot be withdrawn
- C. 'B' RPS half scram signal is generated and control rod 14-19 cannot be withdrawn
- D. Full scram signal is generated and control rod 14-19 is rapidly inserted to 00

Answer:

A is incorrect. With shorting links removed, a non-coincidence logic exists such that any single SRM upscale trip will cause a full reactor scram. Additionally, a rod block would be inserted due to the upscale condition.

B is incorrect. With shorting links removed, a non-coincidence logic exists such that any single SRM upscale trip will cause a full reactor scram.

C is incorrect. With shorting links removed, a non-coincidence logic exists such that any single SRM upscale trip will cause a full reactor scram. Additionally, a rod block would be inserted due to the upscale condition.

D is correct. With shorting links removed, a non-coincidence logic exists such that any single SRM upscale trip will cause a full reactor scram. Control rod 14-19 is rapidly inserted into the core under scram conditions.

K/A: 215004 A3.03

K/A Statement: Ability to monitor automatic operations of the Source Range Monitor (SRM) System including RPS status

References: TM-OP-058-ST Rev. 9
TM-OP-078A-ST Rev. 4

Applicant Ref: None

Learning Objective: TM-OP-058-OB Rev. 0: 2486n/2487b

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.7/45.7

Comments:

New

Question 38

Unit 1 is operating at full power with APRM Channel 1 bypassed due to an INOP fault trip.

An unrelated problem now results in the following condition:

- APRM HI-HI/INOP (AR-103-A06) alarm
- APRM Channel 3 is failed upscale.

Evaluate these conditions to determine the plant response and required procedural response to the event.

- A. An RPS FULL Scram will occur, take action for full scram.
- B. An RPS HALF Scram will occur, bypass the APRM and reset the half scram.
- C. NO RPS Scram Channels trip, bypass the APRM.
- D. NO RPS Scram Channels trip, maintain the APRM tripped to satisfy Technical specifications.

K&A Rating: 215005 K4.02 (4.1/4.2)

K&A Statement: **Knowledge of AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM design feature(s) and/or interlocks which provide for the following: Reactor SCRAM signals**

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not understand the logic configuration when one of the APRM channel is bypassed. When a APRM channel is bypassed the trip logic becomes two out of three instead of two out of four.
- B. **Incorrect but plausible:** Plausible if the applicant does not understand the allowable bypass APRM channels, and when a APRM channel is bypassed the trip logic becomes two out of three instead of two out of four
- C. **Incorrect but plausible:** Plausible if the applicant does not understand the allowable bypass APRM channels. Only one APRM channel is allowed to be bypassed.
- D. **Correct:** No scram will occur because the logic becomes two out of three, and the failed APRM channel needs to be tripped per technical specification.

References: TM-OP-078D-FS, PRNMS, Rev. 05 Applicant Ref: NONE

Learning Objective: TM-OP-078D-OB, Rev. 3, 15716

Question source: SQ Bank.

Question History: None

Cognitive level: Memory/Fundamental knowledge: X

10CFR 41.7

Comments:

Question 39

Unit 1 conditions are as follows:

- Recirc drive flow is 96.0 percent
- Reactor power is 98.0 percent
- An APRM '4' failure causes it to indicate 108 percent power (both ODA bar graphs)
- No self-test faults are detected by APRM '4'
- No APRMs are bypassed

Which one of the following describes the results of the above failure?

- A. Upscale Annunciator/alarm only
- B. Upscale Annunciator/alarm and Rod Block signal to RMCS only
- C. Annunciator/alarm, Rod Block signal to RMCS, and 1 vote to all four voters
- D. Annunciator/alarm, Rod Block signal to RMCS, and 1 vote ONLY to the voter that uses APRM '4' as "Home"

Answer:

APRM STP High Alarm, $(0.55(96.0) + 54.2\%)=107.0\%$: (flow biased, clamped at 108%) - Alarm and rod block only

APRM STP High Trip= $(.55(96.0)+58.7\%)=111.5\%$: (flow biased, clamped at 113.5%) - Alarm, vote and rod block

A is incorrect. Indicated power exceeds STP High Alarm but not STP High Trip, resulting in alarm and rod block to RMCS

B is correct. Indicated power exceeds STP High Alarm but not STP High Trip, resulting in alarm and rod block to RMCS

C is incorrect. Indicated power exceeds STP High Alarm but not STP High Trip, resulting in alarm and rod block to RMCS; no scram votes.

D is incorrect. Indicated power exceeds STP High Alarm but not STP High Trip, resulting in alarm and rod block to RMCS; no scram votes.

K/A: 215005 A3.08

K/A Statement: Ability to monitor automatic operations of the Average Power Range Monitor/Local Power Range Monitor System including control rod block status

Modified from SSES ILO bank question 2450

References: TM-OP-052-ST Rev. 0
EO-000-103 Bases Rev. 8
AR-114-E01 Rev. 23

Applicant Ref: None

Learning Objective: TM-OP-052-OB Rev. 1: 2038n
TM-OP-052-OB Rev. 1: 2037d/f

Question source: Modified

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.7/45.7

Comments:

Modified from SSES bank question 2450

Question 40

Why is the HPCI Test Line Isolation Valve, HV-155-F011 opened during the operator actions of ON-100-009, CONTROL ROOM EVACUATION?

- A. To provide a flowpath for HPCI in CST to CST mode as needed for reactor pressure control.
- B. To ensure minimum flow protection is available for both HPCI and RCIC.
- C. To ensure that the RCIC piping remains filled and vented even if multiple vessel injections are required.
- D. To permit RCIC operation in CST to CST mode for the purpose of increased RPV cooldown.

K&A Rating: 217000 K1.01 (3.5)

K&A Statement: **Knowledge of the physical connections and/or cause-effect relationships between REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) and the following:** Condensate storage and transfer system

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not recall that HPCI cannot be operated from outside the Control Room in pressure control mode.
- B. **Incorrect but plausible:** Plausible if the applicant does not recall that min flow lines do not use this flowpath.
- C. **Incorrect but plausible:** Plausible if the applicant does not recall that keepfill system performs this function.
- D. **Correct:** It provides a flowpath for RCIC in CST to CST.

References: ON-100-009, Rev. 29

Applicant Ref: NONE

Learning Objective: 2018, TM-OP-050-OB, LO Systems, Reactor Core Isolation Cooling

Question source: SQ Bank (Not used in last 2 NRC Exam).

Question History: Not used on 2011 or 2008 SQ Written

Cognitive level: Memory/Fundamental knowledge: X

10CFR 41.2 to 41.9 / 45.7 to 45.8

Comments:

Question 41

Unit 1 conditions as follows:

- Reactor Power is 56%
- RCIC is running for quarterly flow verification
- Control rod pattern adjustments are in progress
- Restoring EHC fluid to #2 TCV following maintenance

While taking data for RCIC, the following conditions now exist:

- AR-106-A12, 125 VDC Panel 1L610 System Trouble, is in alarm
- NPO confirms 125 VDC Distribution Panel 1D614 low voltage
- A reactor scram has just occurred due to EHC pressure swings

What actions are required, if any, if RCIC is needed to mitigate the plant transient?

- A. No actions required, RCIC will auto initiate at -30 inches, and align for injection.
- B. Reset RCIC overspeed, and manually startup RCIC using trip and throttling valve
- C. Perform manual component by component startup and adjust flow controller as required
- D. RCIC will auto initiate at -30 inches but manual control of flow controller will be required

Answer:

A is incorrect. Loss of 1D614 causes a loss of inverter power, causing loss of the turbine control system and causing the turbine governor valve failing to the full open position, causing a turbine overspeed condition.

B is correct. Loss of 1D614 causes a loss of inverter power, causing loss of the turbine control system and causing the turbine governor valve failing to the full open position, causing a turbine overspeed condition. Turbine overspeed condition must be reset, and turbine can be recovered by starting up using the trip and throttling valve.

C is incorrect. Turbine is tripped on overspeed and must be reset; manual component by component startup in this condition would result in a subsequent overspeed.

D is incorrect. Loss of 1D614 causes a loss of inverter power, causing loss of the turbine control system and causing the turbine governor valve failing to the full open position, causing a turbine overspeed condition.

K/A: 217000 A2.05 3.3/3.3

K/A Statement: Ability to (a) predict the impacts of D.C. power loss on the Reactor Core Isolation Cooling System (RCIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations

References: TM-OP-050-ST Rev. 0 Applicant Ref: None

Learning Objective: TM-OP-050-OB Rev. 1: 2015a

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.5/45.6

Comments:

New

Question 42

The following conditions exist following a LOCA:

- Rx vessel water level -130"
- Both ADS Logic timers initiated
- Drywell pressure is at its peak of 1.73 psig.

90 seconds after the timer initiation, Rx vessel water level is at -125" and drywell pressure is at 1.69 psig. What, if any, is the expected operator action related to ADS in accordance with AR-110-001?

- A. Depress ADS Logic Timer Resets to reset timer.
- B. Inhibit ADS in accordance with OP-183-001.
- C. No actions due to automatic ADS reset.
- D. Manual initiate ADS due to failure of auto initiation in accordance with OP-183-001.

K&A Rating: 218000 2.4.31 (4.2/4.1)

K&A Statement: **Knowledge of annunciator alarms, indications, or response procedures.**

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not recall that the low level signal does not seal in and it auto resets above -129".
- B. **Incorrect but plausible:** Plausible if the applicant does not recall that the low level signal does not seal in and it auto resets above -129". Also, inhibiting ADS is not allowed unless directed by emergency operating procedures.
- C. **Correct:** The low level signal does not seal in and it auto resets above -129".
- D. **Incorrect but plausible:** Plausible if the applicant does not recall the initiation setpoint of the ADS logic, which requires low reactor water level (-129") or below for seven minutes.

References: AR-110-001, Rev. 14 Applicant Ref: NONE
 TM-OP-83E-ST, Automatic
 Depressurization System, Rev. 0

Learning Objective: TM-OP-83E-OB, Automatic Depressurization System, Rev. 0, 2105

Question source: New.

Question History: None

Cognitive level: Memory/Fundamental knowledge X
 Comprehension/Analysis:

10CFR 41.10 / 45.3

Comments:

Question 43

During normal power operations, you receive the following annunciator:

- RWCU LEAK DET ISO LOGIC B HI TEMP [AR-101-001 (A03)]

The PCO reports that only the 'B' Channel ambient temperature instrument (Riley Module TSH-G33-1N600B) is upscale (appears failed), but all other instruments are reading normally.

When you perform a walkdown of RWCU, you observe that isolation valves are open and pumps are running normally. Which of the following is the correct diagnosis of this condition?

- A. This is abnormal, since an isolation of both the inboard and outboard valves should have occurred
- B. This is abnormal, since an isolation of the outboard valve should have occurred
- C. This is normal, since only one instrument is reading high
- D. This is normal, since this annunciator comes from the recorder, which is separate from the isolation logic

Answer:

A is incorrect. Applicant may believe that since this is a valid isolation signal that both the inboard and outboard isolation valves will close; most isolations close the F001 Inboard and F004 Outboard valves, steam leak detection does not.

B is correct. For a 'B' isolation logic high temperature, the outboard isolation valve HV-144F004 closes. When HV-144F004 leaves its open position, this also trips the RWCU recirc pumps.

C is incorrect. This is a valid isolation signal. Some of PCIS logic is one out of two taken twice, applicant may incorrectly believe that an 'A' isolation logic is also required.

D is incorrect. This is a valid isolation signal. Some of PCIS logic is one out of two taken twice, applicant may incorrectly believe that an 'A' isolation logic is also required. Alarms are provided, but they are provided at a lower setpoint by separate equipment, and as a warning that an isolation may be forthcoming.

From AR-101-001 (A03):

1. AUTOMATIC ACTION:

IF condition exists for one second:

- 1.1 HV-144F004 RWCU INLET OB ISO **CLOSES**.
- 1.2 WHEN HV-144F004 leaves open position, 1P221A & 1P221B, RWCU RECIRC PUMP TRIP.
- 1.3 At 40 gpm on FICS-14574A or when HV-144F004 leaves open position the Filter Demin Hold Pump 1P223A & 1P223B start and hold valves **OPEN**.

From TM-OP-059B-ST:

Like the MSIVs, RWCU has some specific signals not associated with other systems. These signals are part of the Steam Leak Detection System, which is discussed later. All isolation signals close F001 and F004 unless otherwise noted.

- Reactor Level 2 (-38")
- Manual
- High Delta Flow (59 gpm, 45 sec TD) Compares RWCU flow with the sum of return flow and reject flow.
- High System Flow (462 gpm, 5 sec TD)
- High Area Temperature (See Steam Leak Detection System)
- High Area Delta Temperature (Alarm Function Only-see Steam Leak Detection System)
- Standby Liquid Control Initiation - (F004 only) (Not part of leakage detection.)
- Non-Regenerative Heat Exchanger High Temperature - (F004 only) (Not part of leakage detection.)

STEAM LEAK DETECTION ISOLATION

(Reference Attachment 3, Table 2, Isolation Matrix)

The Steam Leak Detection System limits radioactive releases and inventory losses for leaks outside of the Containment. This Detection and Isolation system deals with individual systems and not multiple systems. It interfaces with NSSSS in some instances for specific systems, but not for total Containment isolations.

The system is comprised of temperature detectors arranged throughout areas where high-energy piping or systems are located. Temperature indicating trip units and temperature recorders are located on 1C614. Reference Figures 22, 23, and 24.

One sensor in any trip system (except for the MSIVs) will cause isolation. Keylock Bypass Switches on 1C614 for each trip system (except for the MSIVs) allow for testing. There is also a one-second time delay for every temperature sensor, so that voltage spikes will not cause an Isolation.

K/A 223002 A1.02 Primary Containment Isolation System/NSSS 3.7/3.7

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operating the Primary Containment Isolation System/Nuclear Steam Supply Shut-Off controls including: Valve closures

References: TM-OP-059B Rev. 7

Applicant Ref: None

Learning Objective: TM-OP-059B-OB Rev. 1: 2120b/2142j

Question source: Modified SSES bank #491

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehension/Analysis:

10CFR55 41.5/45.5

Comments:

Question 44

Unit 1 is operating at 90% power and a loss of power to 125 VDC distribution panel 1D614 occurs. The 125V DC PANEL 1L610 SYSTEM TROUBLE (AR-106-A12) alarm is received in the Control Room.

Which of the following is correct concerning this situation?

- A. ADS SRVs associated manual hand switches on panel 1C601 are functional.
- B. All ADS valves can be remotely operated from Panel 1C631 (Lower Relay Room).
- C. Non-ADS SRVs associated manual hand switches on panel 1C601 are functional.
- D. ADS capability is lost, but the relief mode of all SRV's will still function from panel 1C601.

K&A Rating: 239002 K3.03 (4.3/4.4)

K&A Statement: **Knowledge of the effect that a loss or malfunction of the RELIEF/SAFETY VALVES will have on following:** Ability to rapidly depressurize the reactor.

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant recalls that ADS function is always available and determines that ADS valves will have control from the control room.
- B. **Correct:** ADS initiation signal will energize one or both of the solenoids. One solenoid energizing is all that is needed for ADS operation. Division I is powered from 1D614 and Division II is powered from 1D624. Div I remote operation capability is located in the Upper relay room. If Div I remote capability is lost, Div II capability from the Lower relay room is still available.
- C. **Incorrect but plausible:** Plausible if the applicant does not recall that both ADS and non-ADS functions are lost from the control room panel 1C601.
- D. **Incorrect but plausible:** Plausible if the applicant does not recall that both ADS and non-ADS functions are lost from the control room panel 1C601 but the ADS capability is not lost.

References: TM-OP-083-ST, Main Steam System, Applicant Ref: NONE
Rev. 10.
TM-OP-83E-ST, Automatic
Depressurization System, Rev. 0.

Learning Objective: TM-OP-083E-OB, LO Systems, Automatic Depressurization System
Rev01, 2103

Question source: New.

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehension/Analysis

10CFR 41.7 / 45.4

Comments:

Question 45

Plant conditions are as follows:

- Reactor power is 100%
- Narrow Range Level instrument A (NRLA) has failed upscale
- No operator or maintenance actions are taken for the failed NRLA transmitter
- 4 hours later, Narrow Range Level instrument C (NRLC) transmitter fails upscale

Which of the following describes the response of the feedwater level control system?

- A. Reactor feedwater pumps and the main turbine trip on high reactor water level immediately
- B. Reactor feedwater pumps and the main turbine trip on high reactor water level after a 90 second time delay
- C. Reactor feedwater pumps reduce speed due to the indicated high reactor water level, causing a reactor scram on low level
- D. Reactor feedwater pump speed and reactor water level is unchanged due to selection of an operable level instrument

Answer:

A is incorrect. A high reactor water level trip will not occur due to the automatic removal of these signals from the L8 trip logic 90 seconds after each failure. The trip input is removed without operator action.

B is incorrect. A high reactor water level trip will not occur due to the automatic removal of these signals from the L8 trip logic 90 seconds after each failure. The trip is removed without operator action. If the trip were to occur, it would happen without a time delay.

C is incorrect. With the level transmitters failed upscale, they are declared 'Unusable' by the reactor feedwater control system. This removes them from service, and in the case of NRLA will substitute a low median reactor water level in its place. The system will select an operable level instrument for Selected Level, and RPV Water Level will remain relatively unchanged, RFPT speeds will remain relatively unchanged and a SCRAM will not occur

D is correct. With the level transmitters failed upscale, they are declared 'Unusable' by the reactor feedwater control system. This removes them from service, and in the case of NRLA will substitute a low median reactor water level in its place. The system will select an operable level instrument for Selected Level, and RPV Water Level will remain relatively unchanged

From TM-OP-045I-ST:

The upscale failure of any two level transmitters will result in only one level transmitter being automatically removed from service. The system will select an operable level instrument for Selected Level, and RPV Water Level will remain relatively unchanged

Signals may be removed from service and declared UNUSABLE under the following conditions:

- OOS – The signal must be in range (normally 4 – 20 ma)
 - The signal is out of range high if signal strength > 102 percent of scale (> 20.4 ma) and is automatically removed from service.
- AUTO – If the ICS compound block is placed in manual by NSE or I&C Maintenance, the signal is removed from service.
- BAD – Based on evaluation of raw counts or FBM communications timeout.

RFPT Level 8 trip (+ 54 inches)

NRLA, NRLB, and NRLC provide input signals to RFPT high level trips. The logic is two-out-of-three. If a signal is "BAD or bypassed" it is removed from the L8 trip logic after a 90 second time delay, and the remaining two signals provide the trip using two-out-of-two logic. Only one signal may be automatically removed from service due to excess deviation (± 10 inches).

K/A: 259002 K5.01 Reactor Water Level Control System 3.1/3.1

K/A Statement: Knowledge of the operational implications of Foxboro controller operation as it applies to Reactor Water Level Control System

References: TM-OP-045I-ST Rev. 1

Applicant Ref:
Power/Flow Map
GO-100-009 Att. B

Learning Objective: TM-OP-045I Rev. 1: 16014

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehension/Analysis:

10CFR55 41.5/45.3

Comments:

Question 46

- Unit 1 is in Mode 2; Heatup in progress, with IRMs on Range 8.
- Unit 2 is at rated power with all equipment OPERABLE.
- Unit 2 experiences a LOCA, causing Drywell pressure to peak at 7.5 psig.
- Unit 2 RPV water level dropped to -25 inches, and is now at +35 inches and stable following the event.

Which of the following describes the response of Reactor Building Ventilation System Zones (I, II, II) and the Standby Gas Treatment (SGT) System?

	Zone I	Zone II	Zone III	SGT Fan "A"	SGT Fan "B"
A.	Does <u>NOT</u> Isolate	Does <u>NOT</u> Isolate	Isolates	Starts	Starts
B.	Isolates	Isolates	Does <u>NOT</u> isolate	Starts	Remains Off
C.	Does <u>NOT</u> Isolate	Isolates	Isolates	Starts	Starts
D.	Isolates	Does <u>NOT</u> Isolate	Does <u>NOT</u> Isolate	Remains Off	Starts

K&A Rating: 261000 A3.02 (3.2/3.1)

K&A Statement: **Ability to monitor automatic operations of the STANDBY GAS TREATMENT SYSTEM including: Fan start**

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not recall the setpoint for Zone III isolation signals. Zone III isolates on LOCA U1 at 1.72 psig or -38 inches and LOCA U2 at 1.72 psig or -38 inches.
- B. **Incorrect but plausible:** Plausible if the applicant does not recall the set point for Zone I isolation and SGT fan start logic.
- C. **Correct:** Zone II and III isolate, and both trains of SGT start. Zone I is unaffected. Zone I(II) isolation signals are HI DW Pressure at 1.72 psig on affected Unit, Lo RPV level at -38 inches on affected Unit.
- D. **Incorrect but plausible:** Plausible if the applicant does not recall the set point for Zone I isolation and SGT fan start logic.

References: TM-OP-070 and 034

Applicant Ref: NONE

Learning Objective: TM-OP-070-OB, LO Systems, Standby Gas Treatment System Rev. 02, 1991

Question source: SQ Bank.

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis X

10CFR 41.7/45.7

Comments:

Question 47

Given the following conditions:

- C EDG is out of service due to catastrophic turbocharger failure
- E EDG has not yet been realigned to supply C EDG loads
- At T=0 seconds, a loss of the T-10 line occurs
- At T=15 seconds, a LOCA occurs on unit 1
- All automatic actions have taken place

Based upon the above conditions, describe which Core Spray pumps will start, and when:

- A. CS pumps A, B, and D will start at T=25.5 seconds
- B. CS pumps A, B, and D will start at T=30 seconds
- C. CS pumps A, B, C, and D will start at T=25.5 seconds
- D. CS pumps A, B, C, and D will start at T=30 seconds

Answer:

A is incorrect. All core spray pumps will start. In this case, all have power from offsite for the given conditions and will start with a 15 second time delay. This answer may be chosen if applicant believes that C Core Spray pump has no power and chooses time delay for LOOP.

B is incorrect. All core spray pumps will start. In this case, all have power from offsite for the given conditions and will start with a 15 second time delay. This answer may be chosen if applicant believes that C Core Spray pump has no power, but correct time delay for power available from offsite.

C is incorrect. All core spray pumps will start, however, the applicant may choose this answer using the incorrect time delay. Time delay for this answer is for LOOP.

D is correct. All core spray pumps have power from offsite and will all start with a 15 second time delay

Pump start logic and time delays:

Offsite power available: All Pumps start after 15-second TD

LOOP: Each CS Pump starts 10.5 seconds after respective Diesel Breaker closes

K/A: 262001 K3.01 A.C. Electrical Distribution 3.5/3.7

K/A Statement: Knowledge of the effect that a loss or malfunction of the A.C. Electrical Distribution will have on major system loads

References: TM-OP-004-ST Rev. 2 Applicant Ref: None
TM-OP-051-FS Rev. 0

Learning Objective: TM-OP-004-OB Rev. 2: 2239d/e
TM-OP-051-OB Rev. 2: 2080h

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.7/45.4

Comments:

New

Question 48

Both units are operating at 100% power.

- The crew is performing a surveillance test for full load testing of the 'A' Emergency Diesel Generator (EDG).
- The EDG has just been synchronized to the 4160V 1A bus using SO-024-001A, Monthly Diesel Generator 'A' Operability test.
- 'A' EDG is running at 4000 kW
- Current combined loading of the ESS and Startup buses is 1900 kW.
- 45 minutes into the 1 hour run of the 'A' EDG, a loss of off-site power (LOOP) occurs.

Based on these conditions, which one of the following describes (1) the status of the 'A' EDG and its output breaker with respect to Unit 1 ESS bus 1A, and (2) any manual actions required, if any?

- A. (1) 'A' EDG is running; output breaker 20104 open
(2) EDG output breaker 20104 must be manually closed after resetting the lockout
- B. (1) 'A' EDG is tripped; output breaker 20104 open
(2) EDG 'A' will restart automatically in Emergency Mode and no manual actions required
- C. (1) 'A' EDG is running; output breaker 20104 closed
(2) EDG is running in Non-Emergency Mode and the diesel will have to be manually separated from the bus 1A
- D. (1) 'A' EDG is tripped; output breaker 20104 open
(2) EDG 'A' will restart automatically in Emergency Mode and EDG output breaker 20104 must be manually closed

K&A Rating: 264000 A2.07 (3.5/3.7)

K&A Statement: **Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** Loss of off-site power during full-load testing

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not understand the relationship between LOOP and breaker tripping schemes.
- B. **Incorrect but plausible:** Plausible if the applicant believes that the EDG will trip on overexcitation or underfrequency and the EDG will restart in emergency mode and the output breaker will automatically close.
- C. **Correct:** If a EDG was running in Test Mode when a LOOP occurred, then one of the following happens:
 - 1) The overload condition will cause the EDG to trip on either Overexcitation or Underfrequency. EDG trip will cause output bkr to open. EDG will restart in Emergency Mode, and it will be available to supply power to both ESS buses.
 - 2) Bus voltage will drop sufficiently to operate one of the Bus undervoltage schemes. The output bkr will open, the EDG will swap to the emergency mode, and it will be available to supply power to both ESS buses.
 - 3) If the load on the ESS and Startup Buses connected to the Diesel is within its' load capability, the Diesel continues to run in the Non-Emergency Mode supplying the Buses. The Diesel will have to be manually separated from the Buses by either opening the Output Breaker to the ESS Bus, or by tripping the Diesel. **(This will occur due to the load on the ESS and Startup Buses (1900 kW) are within EDG's load capability (4000 kW)).**
- D. **Incorrect but plausible:** Plausible if the applicant believes that the EDG will trip on overexcitation or underfrequency and the EDG will restart in emergency mode, but manual action will be required to close the EDG output breaker.

References: TM-OP-024-ST, Emergency Diesel Generators A-D, Rev. 10 Applicant Ref: NONE

Learning Objective: TM-OP-024-OB, LO Systems, Emergency Diesel Generators A - D Rev01, 2260

Question source: New.

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis X

10CFR 41.5/45.6

Comments:

Question 49

Given the following conditions:

- Unit 1 is at 100% power
- Battery charger 1D684 has been removed from service due to failure of the rectifier bank
- Battery charger 1D683 output breaker has just tripped

10 hours have elapsed. Which of the following describes an instrument impacted by these conditions?

- A. Source Range Channel 'A'
- B. RBCCW Radiation Monitor
- C. Source Range Channel 'D'
- D. Offgas Linear Radiation Monitor

Answer:

A is incorrect. This load is powered from 1D672 via battery 1D670 and chargers 1D673/1D674. May be chosen if applicant cannot recall which Source Range division is powered by these components.

B is incorrect. This load is powered from 1D672 via battery 1D670 and chargers 1D673/1D674. May be chosen if applicant cannot recall which battery and chargers supply power to the RBCCW radiation monitor

C is correct. This load is supplied by 1D682 via battery 1D680 and battery chargers 1D683/1D684.

D is incorrect. This load is powered from 1D672 via battery 1D670 and chargers 1D673/1D674. May be chosen if applicant cannot recall which Source Range division is powered by these components.

From TM-OP-075-ST:

Load Center 1D672

- RBCCW Radiation Monitor
- Offgas Pretreatment Linear Radiation Monitor
- Division I SRMs, IRMs and Trip Auxiliary Units

Load Center 1D682

- Service Water Effluent Radiation Monitor
- Division II SRMs, IRMs and Trip Auxiliary Units

K/A 263000 A1.01 D.C. Electrical Distribution 2.5/2.8

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operating the D.C. Electrical Distribution controls including battery charging/discharging rate

References: TM-OP-075-FS Rev. 0

Applicant Ref: None

Learning Objective: TM-OP-075-OB Rev. 0: 1439b/c

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehension/Analysis:

10CFR55 41.5/45.5

Comments:

New, similar to SQ 2011 question 55

Question 50

Unit One is operating at rated power when an air leak occurs on the common header downstream of the instrument air receivers. The 1A compressor is in LEAD and the 1B compressor is in STANDBY.

- Prior to the leak, pressure was 101 psig being maintained by 1A compressor
- Air pressure decreased to 80 psig over twenty minutes.
- Air pressure then recovered to 85 psig.

Which one of the following statements describes the expected status of the instrument air system at this time?

- A. Both instrument air compressors are running fully loaded and the SA cross-tie pressure control valve is controlling pressure.
- B. Both instrument air compressors are running fully loaded, the Unit 1 to Unit 2 IA cross-tie pressure control valves are open controlling pressure.
- C. The 1A instrument air compressor only is running, the SA cross-tie pressure control valve is open.
- D. The 1A instrument air compressor only is running, the Unit 1 to Unit 2 cross-tie pressure control valve is controlling pressure.

K&A Rating: 300000 K4.02 (3.0)

K&A Statement: **Knowledge of (INSTRUMENT AIR SYSTEM) design feature(s) and or interlocks which provide for the following:** Cross-over to other air systems.

Justification:

- A. **Correct:** PCV-12560 crosstie from service air starts to open when instrument air header pressure is approximately 95 psig, and is fully open at 90 psig. The leak would cause both instrument air compressors to run and service air cross tie to open fully and recover the pressure.
- B. **Incorrect but plausible:** Plausible if the applicant does not understand the system design and how service air provides the back up automatically. The cross tie from Unit 2 is a manual operation that is considered infrequent operation and requires OP-118-002 to be implemented.
- C. **Incorrect but plausible:** Plausible if the applicant does not understand that both air compressors would start and at approximately 95 psig crosstie would start to open.
- D. **Incorrect but plausible:** Plausible if the applicant does not understand the system design and how service air provides the back up automatically. The cross tie from Unit 2 is a manual operation that is considered infrequent operation and requires OP-118-002 to be implemented.

References: TM-OP-018

Applicant Ref: NONE

Learning Objective: 1769, TM-OP-018-OB, LO Systems, Instrument Air Rev0

Question source: SQ Bank.

Question History: Not used on 2011 or 2008 SQ
Written

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis X

10CFR 41.7

Comments:

Question 51

Units 1 and 2 are operating at 100% power with 'A' EDG running loaded for post maintenance testing. Which of the following describes the effect of taking the 'A' EDG voltage regulator switch to the LOWER position?

'A' EDG operating disconnected
from the grid

'A' EDG operating in parallel
with the grid

A. 'A' EDG output voltage remains the same
KVARs lower

Bus voltage rises
KVARs remain the same

B. 'A' EDG output voltage remains the same
KVARs lower

Bus voltage remains the same
KVARs lower

C. 'A' EDG output voltage lowers
KVARs remain the same

Bus voltage remains the same
KVARs lower

D. 'A' EDG output voltage lowers
KVARs remain the same

Bus voltage lowers
KVARs remain the same

Answer: C

A is incorrect. While disconnected from the grid, EDG output voltage will lower and KVAR are solely dependent upon bus loads. While paralleled with the grid, bus voltage will remain unchanged and KVAR will lower.

B is incorrect. While disconnected from the grid, EDG output voltage will lower and KVAR are solely dependent upon bus loads. While paralleled with the grid, bus voltage will remain unchanged and KVAR will lower.

C is correct. While disconnected from the grid, EDG output voltage will lower and KVAR are solely dependent upon bus loads. While paralleled with the grid, bus voltage will remain unchanged and KVAR will lower.

D is incorrect. While disconnected from the grid, EDG output voltage will lower and KVAR are solely dependent upon bus loads. While paralleled with the grid, bus voltage will remain unchanged and KVAR will lower.

K/A: 264000 A4.01 Emergency Generators 3.3/3.4

K/A Statement: Ability to manually operate and/or monitor in the control room: adjustment of exciter voltage

References: TM-OP-024-ST Rev. 10

Applicant Ref: None

Learning Objective: TM-OP-024-OB Rev. 1: 2256d

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.7/45.5 to 45.8

Comments:

New

Question 52

A control rod drive mechanism removal from the reactor core for repair can be accomplished with coolant in the RPV.

What feature allows this evolution to be accomplished?

- A. Installation of a flange on the guide tube prior to removing the internals of the drive mechanism
- B. The thermal shield seating on the hole created when the piston tube is removed from the guide tube
- C. The ball check valve internal to the guide tube housing will seat with reactor pressure to prevent leakage
- D. The metal-to-metal seal formed when the control rod blade sits on a machined surface within the guide tube

K&A Rating: 201003 K1.04 (3.0)

K&A Statement: **Knowledge of the physical connections and/or cause effect relationships between CONTROL ROD AND DRIVE MECHANISM and the following:** Reactor vessel.

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not understand the installation purpose of the flange on guide tube. The Flange cannot be installed until after the removal of the drive.
- B. **Incorrect but plausible:** Plausible if the applicant does not understand the purpose of the thermal shield. The thermal shield is provided to minimize heat transfer to the drive.
- C. **Incorrect but plausible:** Plausible if the applicant does not understand the physical location of the ball check valve. The ball check valve is integral to the guide tube and is removed when the CRDM is removed.
- D. **Correct:** The velocity limiter seals against the vessel surface to allow for drive removal.

References: TM-OP-055B

Applicant Ref: NONE

Learning Objective: 10066, TM-OP-055B-OB, LO Systems, Control Rod Drive Mechanisms Rev02

Question source: SQ Bank.

Question History: Not used on 2011 or 2008 SQ Written

Cognitive level: Memory/Fundamental knowledge: X

10CFR 41.2 to 41.9 / 45.7 to 45.8

Comments:

Question 53

Given the following:

- Unit 1 at 100 percent power
- 1B RBCCW Pump in Run
- 1A RBCCW pump in Standby
- Power is simultaneously lost to 1B216 and 1B237 Motor Control Centers

If power is then restored to 1B237 within 5 minutes, followed by 1B216 two minutes later, which of the following describes the response of the RBCCW system?

- A. Both 1A and 1B RBCCW pumps will be running
- B. 1B RBCCW pump will be running with 1A RBCCW pump in Standby
- C. 1B RBCCW pump must be restarted and 1A RBCCW restored to Standby
- D. Only 1A RBCCW pump will be running in Standby

Answer:

A is incorrect. Applicant may choose this if the low pressure switch input to standby pump logic and its effect on pump restoration sequence is not understood.

B is correct. 1B RBCCW pump will restart upon restoration of power to 1B237. Two minutes allows sufficient system flow and pressure to build, allowing the low pressure switch to reset. Subsequent restoration of power to the Standby 1A RBCCW pump after the pressure switch is reset will cause it to remain in Standby.

C is incorrect. Applicant may choose this if they do not understand a misconception of pump behavior upon restoration of power. RBCCW pumps restart upon restoration of power, whereas TBCCW pumps do not.

D is incorrect. Applicant may choose this if they do not remember correct pump power supplies and pump response to restoration of power.

PUMP	POWER SOURCE	BKR LOCATION
1P210A	1B216	Reactor Building Elevation 683'
1P210B	1B237	Reactor Building 670' elevation

K/A: 400000 K6.05 Component Cooling Water System 3.0/3.1

K/A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: Pumps

References: TM-OP-014-ST Rev. 3

Applicant Ref: None

Learning Objective: TM-OP-014-OB Rev. 0: 10255/1694a/1676b

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.7/45.7

Comments:

New

Question 54

While at rated power, Unit 1 had a Loss of Coolant Accident (LOCA). The reactor scrammed, and the turbine tripped on reverse power. HPCI initiated on high drywell pressure, and is being used to control RPV level. Unit 1 conditions are as follows:

Reactor Pressure is now 320 psig lowering slowly
Reactor Level reached -40 inches and is now 37 inches steady
Containment Pressure is 12 psig
Containment Temperature is 254 °F

With the current plant conditions, what is the status of the Reactor Recirculation Pumps?

- A. Running at 30 percent speed.
- B. Tripped with RPT breakers CLOSED and Recirculation MG Drive Motors tripped.
- C. Tripped with RPT breakers OPEN and Recirculation MG Drive Motors tripped.
- D. Tripped with their Recirculation Pump discharge and Discharge Bypass Valves closed.

Question 55

Given the following:

- An electric ATWS exists on Unit 2
- Reactor power is 43%
- The PCOM has been directed to drive control rods in accordance with EO-200-113 sheet 2, Control Rod Insertion
- Prior to taking any action, the Rod Worth Minimizer experiences a fatal software error

Which of the following describes (1) the impact of this failure and (2) the action required by the PCOM in accordance with EO-200-113 to insert control rods?

- A. (1) Control rod insertion is prevented
(2) Bypass the RWM and depress the Insert Rod pushbutton
- B. (1) Control rod insertion is prevented
(2) Bypass the RWM and depress the Continuous Insert pushbutton
- C. (1) Control rod insertion is NOT prevented
(2) Depress the Insert Rod pushbutton
- D. (1) Control rod insertion is NOT prevented
(2) Depress the Continuous Insert pushbutton

Answer:

A is incorrect. RWM must be bypassed due to insert and withdrawal rod blocks; EO-200-113 guidance directs depressing the Continuous Insert for each of the control rods that are not fully inserted.

B is correct. EO-200-113 directs bypassing the RWM and depressing the Continuous Insert pushbutton for each of the control rods that are not fully inserted.

C is incorrect. Candidate may believe that because reactor power is above the LPAP and LPSP that no blocks will be enforced. RWM must be bypassed due to insert and withdrawal rod blocks; EO-200-113 guidance directs depressing the Continuous Insert for each of the control rods that are not fully inserted.

D is incorrect. Candidate may believe that because reactor power is above the LPAP and LPSP that no blocks will be enforced. RWM must be bypassed due to insert and withdrawal rod blocks; EO-200-113 guidance directs depressing the Continuous Insert for each of the control rods that are not fully inserted.

K/A Statement: Ability to (a) predict the impacts of power supply loss on the Rod Worth Minimizer System (RWM); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations

References: TM-OP-031D-ST Rev. 5

Applicant Ref: None

Learning Objective: TM-OP-031D-OB Rev. 2: 1575a

Question source: Peach Bottom 1/11

Question History: None

Cognitive level: Memory/Fundamental knowledge:

Comprehension/Analysis: X

10CFR55 41.5/45.6

Comments:

New to Susquehanna, taken from Peach Bottom 1/11 exam, question 34.

Question 56

The following conditions exist on Unit 1:

- A transient occurred that has caused Suppression Pool Level to rise.
- Attempts to Lower Suppression Pool level in accordance with OP-159-001 Suppression Pool Cleanup System were unsuccessful.
- HPCI and RCIC were placed on MIN Flow at 25 feet in the Suppression Pool.
- At 35 feet in the Suppression Pool level, Rx SCRAM was performed IAW ON-100-101 and EO-100-102, RPV control was entered at step RC-1 .
- Suppression Pool Level is now 35 feet and rising.

In accordance with EO-100-103, "PC Control", which one of the following is correct?

- A. Transfer HPCI Suction from CST to Suppression Pool
- B. Isolate HPCI and RCIC due to water level exceeding the exhaust lines
- C. Open ALL BPV's to depressurize the RPV
- D. Isolate HPCI only due to water level exceeding the exhaust lines

K&A Rating: 216000 G2.4.20 (3.8/4.3)

K&A Statement: **Knowledge of the operational implications of EOP warnings, cautions, and notes.**

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not recall that the transfer of HPCI suction from CST to suppression pool is aligned when HPCI is injecting into the RPV. HPCI is placed in minflow.
- B. **Incorrect but plausible:** Plausible if the applicant does not recall that if the turbines are already running as pool level exceeds 25', continued operation will not result in adverse consequences. (water entering the exhaust lines resulting in high exhaust pressure during system start)
- C. **Correct:** If rapid depressurization is anticipated and the reactor is shutdown on control rods, Main Turbine Bypass Valves may be used to depressurize the RPV until the rapid depressurization limit, 38 feet is reached. Permission to used Main Turbine Bypass valves is given in override located at RC/P-3 of RPV control.
- D. **Incorrect but plausible:** Plausible if the applicant does not recall that HPCI is isolated ONLY for low suppression pool level concern, due to the concern of exhaust Sparger uncovering (17 ft) will cause the suppression chamber pressure to increase and possibly cause the primary containment to fail.

References: EO-000-103, Rev. 8
AR-114-001, Rev. 23

Applicant Ref: NONE

Learning Objective:

Question source: New.

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehension/Analysis

10CFR 41.10 / 43.5 / 45.13

Comments:

Question 57

- Unit 1 is starting up after a 6 month outage
- Reactor is critical in the source range
- RPV temperature is 126 °F
- RWCU is in letdown mode in automatic
- Feedwater Startup Bypass Valve HV-10640 is currently closed and non-responsive; troubleshooting in progress
- The running CRD pump trips

Which of the following describes the response of reactor water level to these conditions?

- A. Rise due to decay heat generation
- B. Rise due to reactor heat generation
- C. Lower to 30" and stabilize
- D. Lower to -38" and stabilize

Answer:

A is incorrect. With the reactor shutdown for 6 months, little to no decay heat exists in the reactor. With the loss of water addition from the CRD pump, reactor water level will lower due to letdown flow.

B is incorrect. The reactor is critical in the source range. With the loss of water addition from the CRD pump, reactor water level will lower due to letdown flow. Once the reactor reaches the point of adding heat (POAH) at approximately 1% reactor power, reactor water level will rise due to reactor heat generation.

C is correct. With the reactor water cleanup (RWCU) system in letdown mode, the RWCU letdown flow regulating valve HV-144-F033 is controlled automatically using the selected reactor water level. If RPV water level falls to the LLA setpoint (30 inches) the controller ramps to zero percent output at 50 percent/minute. When level recovers to above 39 inches, the controller high output limit is returned to 100 percent and the controller opens the valve as required by the level error signal.

D is incorrect. With the loss of water addition from the CRD pump, reactor water level will lower due to letdown flow. The lowering trend will be stopped by the closing of HV-144-F033 RWCU letdown flow regulating valve at 30" RPV water level. Water level will not reach the RWCU system isolation setpoint of -38" RPV water level.

K/A: 204000 A4.08

K/A Statement: Ability to manually operate and/or monitor in the control room: reactor water level

References: TM-OP-061-ST Rev. 9

Applicant Ref: None

Learning Objective: TM-OP-061-OB Rev. 1: 1693L

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:

Comprehension/Analysis: X

10CFR55 41.7/45.5 to 45.8

Comments:

Question 58

Given:

- A LOCA has occurred in the Drywell.
- RPV level is +20 inches and is steady.
- Suppression Chamber pressure is 7.3 psig and rising slowly.
- The CRS directs "B" RHR be placed in Suppression Chamber Spray.

For the above conditions, which operator actions and plant conditions are required to establish Suppression Chamber Spray?

LPCI injection valve, F017B must be (1). Suppression Chamber Spray valves F027B and F028B must be opened, by (2) the initiation signal.

- A. (1) Opened (2) overriding
- B. (1) Opened (2) resetting
- C. (1) Closed (2) overriding
- D. (1) Closed (2) resetting

K&A Rating: 230000 A4.06 (4.0/3.9)

K&A Statement: **Ability to manually operate and/or monitor in the control room:** Valve logic reset following automatic initiation of LPCI/RHR in injection mode.

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not recall the suppression chamber spray initiation logic. F017B must be closed to permit F027B and F028B to be opened.
- B. **Incorrect but plausible:** Plausible if the applicant does not recall the suppression chamber spray initiation logic. F017B must be closed to permit F027B and F028B to be opened. Also with the present plant conditions the initiation signal can not be reset.
- C. **Correct:** F017B must be closed to permit opening F027B and F028B. LOCA initiating signal must be present to override valves.
- D. **Incorrect but plausible:** Plausible if the applicant does not recall that with the present plant conditions the initiation signal can not be reset.

References: TM-OP-049-ST RHR, Rev. 07 Applicant Ref: NONE

Learning Objective: 181, TM-OP-049-OB, LO Systems, Residual Heat Removal, Rev. 02

Question source: New.

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis X

10CFR 41.7 / 45.5 to 45.8

Comments:

Question 59

Unit 1 is in MODE 4

- The 'A' Loop of RHR is in Suppression Pool Cooling using 1A RHR pump
- The 'B' Loop of RHR is in Shutdown Cooling using 1D RHR pump
- A fault on the 1A201 bus causes actuation of the Bus Primary Differential Relay

Which one of the following correctly describes the effect of this event?

The ___(1)___ EDG starts and _____(2)_____

- A. (1) A
(2) runs unloaded, resulting in a loss of Suppression Pool Cooling
- B. (1) D
(2) runs unloaded, resulting in a loss of Shutdown Cooling
- C. (1) A
(2) loads the 1A ESS bus, restoring available power for Suppression Pool Cooling
- D. (1) D
(2) loads the 1D ESS bus, restoring available power for Shutdown Cooling

Answer:

A is correct. 'A' EDG powers the 1A201 bus. It will receive a start signal for low bus voltage, but will be unable to close its output breaker to power the bus due to the differential overcurrent lockout of the bus.

B is incorrect. 'D' EDG powers the 1A204 bus and will not receive a start signal for this bus fault. Applicant may choose this if they do not remember the correct power supplies for RHR pumps.

C is incorrect. 'A' EDG powers the 1A201 bus. It will receive a start signal for low bus voltage, but will be unable to close its output breaker to power the bus due to the differential overcurrent lockout of the bus. Applicant may choose this if they do not understand the effect of a bus lockout on EDG and 4kV bus transfer operations

D is incorrect. 'D' EDG powers the 1A204 bus. It will receive a start signal for low bus voltage, but will be unable to close its output breaker to power the bus due to the differential overcurrent lockout of the bus. Applicant may choose this if they do not remember the correct pump power supplies, and the effect of a bus lockout on EDG and 4kV bus transfer operations

K/A: 219000 K2.02 RHR/LPCI: Suppression Pool Cooling Mode 3.1/3.3

K/A Statement: Knowledge of electrical power supplies to the following: Pumps

References: TM-OP-049-ST Rev. 7 Applicant Ref: None
AR-016-001 Rev. 44
TM-OP-024-ST Rev. 10

Learning Objective: TM-OP-049-OB Rev. 0: 10499a

Question source: Modified Peach Bottom 12/08

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.7

Comments:

Question 60

While performing Main Turbine shell warming in accordance with OP-193-001 "Main Turbine Operation" the operator is cautioned to ensure turbine first stage pressure remains below 130.1 psig. The reason for this caution is to prevent _____".

- A. rolling the main turbine off the turning gear
- B. differential expansion between the turbine shell and rotor
- C. damaging the seats in the moisture separator dump valves
- D. exceeding the setpoint for the turbine stop valve and control valve scram bypass

K&A Rating: 245000 K5.02 (2.8/3.1)

K&A Statement: **Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS:** Turbine operation and limitations

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not recall that lift pumps are secured to prevent rolling the main turbine off the turning gear.
- B. **Incorrect but plausible:** Plausible if the applicant does not recall that differential expansion concerns are addressed by the pre-warming direction in OP-193-001.
- C. **Incorrect but plausible:** Plausible if the applicant does not recall that any delay in restoring feedwater heating between the completion of turbine roll and the restoration of level control will have adverse effect on the moisture separator dump valves.
- D. **Correct:** ON-193-001 caution states that “Do not allow HP Turbine pressure to increase to 130.1 PSIG on TEP02 or TEP03 OR Control and Stop Valve Closure Scram Bypass is removed AND a Reactor Scram will occur.”

References: OP-193-001, Rev. 38

Applicant Ref: NONE

Learning Objective: 1611, TM-OP-093-OB, LO Systems, Main Turbine Rev. 01

Question source: New.

Question History: None

Cognitive level: Memory/Fundamental knowledge: X

10CFR 41.5/45.3

Comments:

Question 61

Which of the following describes the normal makeup source to the Unit 1 and Unit 2 Spent Fuel Pools?

	<u>Unit 1</u>	<u>Unit 2</u>
A.	Condensate Transfer System	Condensate Transfer System
B.	Demineralized Water System	Condensate Transfer System
C.	Condensate Transfer System	Demineralized Water System
D.	Demineralized Water System	Demineralized Water System

Answer:

A is incorrect. Unit 1 Spent Fuel Pool normal makeup is from the Condensate Transfer System. Unit 2 Spent Fuel Pool normal makeup is from the Demineralized Water System.

B is incorrect. Unit 1 Spent Fuel Pool normal makeup is from the Condensate Transfer System. Unit 2 Spent Fuel Pool normal makeup is from the Demineralized Water System.

C is correct. Unit 1 Spent Fuel Pool normal makeup is from the Condensate Transfer System. Unit 2 Spent Fuel Pool normal makeup is from the Demineralized Water System.

D is incorrect. Unit 1 Spent Fuel Pool normal makeup is from the Condensate Transfer System. Unit 2 Spent Fuel Pool normal makeup is from the Demineralized Water System.

K/A: 233000 K4.06 2.9/3.2

K/A Statement: Knowledge of Fuel Pool Cooling and Clean-Up design feature(s) and/or interlocks which provide for the following: Maintenance of adequate pool level

References: TM-OP-035-ST Rev. 4

Applicant Ref: None

Learning Objective: TM-OP-035-OB Rev. 1: 2204b, 2200e/l

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehension/Analysis:

10CFR55 41.7

Comments:

Question 62

Both Units are at rated power. A fire causes a loss of 125V DC distribution panel 1D625 and also the Fire Water header pressure lowers to 90 psig. Assuming no operator actions, what effect, if any, will this have on the Fire Water System?

- A. ONLY the Motor Driven Fire Pump will start.
- B. ONLY the Diesel Driven Fire Pump will start.
- C. The Motor Driven and Diesel Driven Fire Pumps will start.
- D. Neither the Motor Driven nor Diesel Driven Fire Pumps will start.

K&A Rating: 286000 K4.02 (3.3/3.5)

K&A Statement: **Knowledge of FIRE PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following:** Automatic system initiation

Justification:

- A. **Correct:** Only the motor driven fire pump will auto start due to the pressure dropping below 95 psig. The diesel driven fire pump would auto start when the pressure dropped below 85 psig. 125V DC distribution panel 1D625 is the alternate power supply to FP system (1D615 is normal power supply). Loss of alternate power supply will not disable the auto start of the fire pumps.
- B. **Incorrect but plausible:** Plausible if the applicant does not recall the setpoint for the diesel fire pump auto start.
- C. **Incorrect but plausible:** Plausible if the applicant does not recall the setpoint for the diesel fire pump auto start.
- D. **Correct:** Plausible if the applicant does not recall that 1D625 is the alternate power supply, the normal power supply (1D615) is still available.

References: TM-OP-013-ST, Rev. 08.doc Applicant Ref: NONE

Learning Objective: 2295, TM-OP-013-OB, LO Systems, Fire Protection Rev01

Question source: New.

Question History: None

Cognitive level: Memory/Fundamental knowledge
Comprehension/Analysis: X

10CFR 41.7 / 41.3, 45.5 to 45.8

Comments:

Question 63

Given the following conditions:

- Unit 1 is at 100% reactor power
- 'C' Condensate pump trips on overcurrent
- During the transient, condensate header pressure lowered to approximately 265 psig for 12 seconds before recovering

Which of the following describes the status of the reactor feed pumps?

- A. 'A', 'B', 'C' reactor feed pumps in service, no feed pumps tripped
- B. 'B', 'C' reactor feed pumps in service, 'A' reactor feed pump tripped
- C. 'A', 'B' reactor feed pumps in service, 'C' reactor feed pump tripped
- D. 'A' reactor feed pump in service, 'B', 'C' reactor feed pumps tripped

Answer:

A is incorrect. 'C' reactor feed pump trips after a 5 second delay with feed pump suction pressure less than 288 psig, 'B' reactor feed pump after a 15 second time delay, and 'A' reactor feed pump after a 30 second time delay.

B is incorrect. 'C' reactor feed pump trips after a 5 second delay with feed pump suction pressure less than 288 psig, 'B' reactor feed pump after a 15 second time delay, and 'A' reactor feed pump after a 30 second time delay.

C is correct. 'C' reactor feed pump trips after a 5 second delay with feed pump suction pressure less than 288 psig, 'B' reactor feed pump after a 15 second time delay, and 'A' reactor feed pump after a 30 second time delay.

D is incorrect. 'C' reactor feed pump trips after a 5 second delay with feed pump suction pressure less than 288 psig, 'B' reactor feed pump after a 15 second time delay, and 'A' reactor feed pump after a 30 second time delay.

K/A: 259001 K1.05 Reactor Feedwater 3.2/3.2

K/A Statement: Knowledge of the physical connections and/or cause-effect relationships between Reactor Feedwater System and the following: Condensate system

References: TM-OP-045H-ST Rev. 2

Applicant Ref: None

Learning Objective: TM-OP-045H-OB Rev. 1: 16094j/o, 16099b/c

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehension/Analysis:

10CFR55 41.2 to 41.9/45.7 to 45.8

Comments:

Question 64

With respect to power supply and ability to control from the Unit 1 Remote Shutdown Panel, which of the following is correctly associated with RHRSW Pump 1B?

	<u>Power Supply</u>	<u>RSDP Control</u>
A.	4 kV ESS 1C	NO
B.	4 kV ESS 1C	YES
C.	4 kV ESS 1D	NO
D.	4 kV ESS 1D	YES

K&A Rating: 2.1.30 (4.4/4.0)

K&A Statement: **Ability to locate and operate components, including local controls.**

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not recall that the correct power supply for the 1B RHRSW Pump is ESS 1D and can be controlled from the RSP.
- B. **Incorrect but plausible:** Plausible if the applicant does not recall that the RHRSW Pump 1B is powered from ESS 1D.
- C. **Incorrect but plausible:** Plausible if the applicant does not recall that the 1B RHRSW pump can be controlled from the RSP.
- D. **Correct:** 1B RHRSW pump is powered from ESS 1D and can be controlled from the RSP.

References: Applicant Ref: NONE

Learning Objective: 2061, TM-OP-016-OB, LO Systems, RHR Service Water Rev01

Question source: SQ Bank Modified

Question History: None

Cognitive level: Memory/Fundamental knowledge: X

10CFR 41.7/45.7

Comments:

Question 65

Which of the following describes the operational implications of a failure of the Makeup Outside Air Dampers FD-07551A2/B2 to close during a Secondary Containment Isolation (Zone 3) initiation signal?

- A. The excess airflow through the Makeup Outside Air Dampers will result in lower available airflow to the Cooling Outside Air Dampers, resulting in potential overtemperature and fire in the High Efficiency Charcoal Adsorber Filter
- B. The excess airflow could result in a duct overpressure condition, potential failure of the Standby Gas Treatment System ductwork, and airborne radioactivity hazard
- C. The excess airflow may result in insufficient humidity for proper High Efficiency Charcoal Adsorber function, causing possible higher iodine pass-through and higher offsite post-LOCA doses
- D. The excess air flow will result in the inability of the Standby Gas Treatment System to draw down the Secondary Containment within the required time and higher offsite post-LOCA doses

Answer:

A is incorrect. Overtemperature or fire in the High Efficiency Charcoal Adsorber (HECA) filter provides no input to these dampers, instead provides an input into the crosstie and outside air cooling dampers.

B is incorrect. The dampers being open will not cause a failure of the system ductwork.

C is incorrect. There is a requirement to maintain High Efficiency Charcoal Adsorber (HECA) inlet humidity <70% to ensure proper operation; this is controlled by use of the Main Air Stream Heaters to preheat and reduce the relative humidity of the incoming air. There is no requirement for minimum humidity

D is correct. The Makeup Outside Air Dampers are closed upon system initiation, and allowed to modulate after a 140 second time delay. This delay is required to guarantee the damper will remain closed during the Secondary Containment draw-down period; to assure a differential pressure of greater than -0.25 inches WG is achieved within 125 seconds for Zones I, II, and III, 117 seconds for Zones I, and III, and 118 seconds for Zones II, and III. In order to limit the offsite dose during post-LOCA accident conditions, the dampers need to remain closed, so that the SGTS can "draw down" the Secondary Containment within a three-minute time period. After the initial "draw down" period the damper is allowed to modulate to provide for proper system control.

K/A: 288000 K5.02 Plant Ventilation 3.2/3.4

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to Plant Ventilation Systems: Differential Pressure control

References: TM-OP-070-ST Rev. 5

Applicant Ref: None

Learning Objective: TM-OP-070-OB Rev. 2: 1991a/c

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehension/Analysis:

10CFR55 41.7/45.4

Comments:

Question 66

Unit 1 is operating at 100% power when an unexpected transient occurs.

Which one of the following describes a resulting condition that will violate a Technical Specification Safety Limit?

- A. Drywell pressure rises to 70 psig
- B. Reactor level drops to -165 inches
- C. Reactor steam dome pressure rises to 1320 psig
- D. Minimum Critical Power Ratio (MCPR) lowers to 1.15

K&A Rating: 2.2.22 (4.0/4.7)

K&A Statement: **Knowledge of limiting conditions for operations and safety limits.**

Justification:

- A. **Incorrect but plausible:** Drywell pressure is not a safety limit but plausible since it exceeds the maximum design pressure for the drywell.
- B. **Correct:** Although the limit is applicable when shutdown, the plant will scram on decreasing level and be shutdown. -160 inches level indicated is the safety limit. Reactor water level is required to be above the top of the active fuel (-160") to provide core cooling capability when the reactor is shutdown.
- C. **Incorrect but plausible:** Reactor pressure rising to 1320 psig does not exceed a Safety Limit but plausible since it is above the Safety valve setpoint
- D. **Incorrect but plausible:** Incorrect Plausible since MCPR was not less than the Safety Limit Critical Power Ratios of 1.09.

References: Tech Spec. 2.1.1, 2.1.2, 2.2

Applicant Ref: NONE

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X

10CFR 41.5 / 43.2 / 45.2

Comments:

Question 67

Select the set of primary containment conditions that would exceed a primary containment design limit.

	<u>Drywell Pressure (psig)</u>	<u>SC Pressure (psig)</u>	<u>Drywell Temp (°F)</u>
A.	48.0	22.1	230.6
B.	10.5	16.1	275.1
C.	52.4	48.3	330.8
D.	-4.8	-2.4	125.4

Answer:

A is incorrect. Applicant may choose this because they believe they are exceeding design DW floor DP in the downward direction.

B is correct. $16-10=6$ psid being applied to the DW floor in the upward direction, exceeding the design upward pressure of 5.5 psid

C is incorrect. Applicant may choose this because they believe they are exceeding design max internal primary containment pressure or design drywell temperature.

D is incorrect. Applicant may choose this because they believe they are exceeding max external design pressure

Primary containment T.S. design features:

Max internal pressure: 53 psig

Max external pressure: 5 psig

Max DW floor DP: 28 psid downward
5.5 psid upward

Design temperature: 340 deg F

K/A: 2.1.32 3.8/4.0

K/A Statement: Ability to explain and apply system limits and precautions

References: TM-OP-059-ST Rev. 1
TM-OP-059-FS Rev. 0

Applicant Ref: None

Learning Objective: TM-OP-059-OB Rev. 1: 10357a

Question source: Modified

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.10/43.2/45.12

Comments:

Modified SSES bank question 1998

Question 68

During a declared emergency, a RB NPO must enter an area of the Reactor Building to locate and isolate a leak. The general area radiation level is 3 Rem/hr.

The NPO, age 31, has the following radiation history:

- 1850 mRem cumulative exposure for the current year (TEDE)
- 18 Rem lifetime exposure to this date (TEDE)
- No dose extensions have been obtained
- NRC form 4 completed and on file

The NPO has been given 45 minutes to complete the task.

Which one of the following radiation exposure limits, if any, would be exceeded if the NPO performs this task?

- A. No exposure limits would be exceeded
- B. Administrative Dose Control limits
- C. NRC Exposure limit
- D. Administrative Dose Control AND NRC Exposure limits

K&A Rating: 2.3.4 (3.2/3.7)

K&A Statement: **Knowledge of radiation exposure limits under normal or emergency conditions.**

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not recall the Administrative Dose control limit of 2000 mr/hr.
- B. **Correct:** $3000 \text{ mr/h} \times .75 = 2250 \text{ mr/hr} + 1850 \text{ mr/hr} = 4100 \text{ mr/hr}$. Result of 4100 mr/hr is above the Administrative Dose control limit of 2000 mr/hr, but below the NRC exposure limit of 5000 mr/hr.
- C. **Incorrect but plausible:** Plausible if the applicant does not recall NRC exposure limit of 5000 mr/hr
- D. **Incorrect but plausible:** Plausible if the applicant does not recall Administrative Dose control limit of 2000 mr/hr and NRC exposure limit of 5000 mr/hr.

References: NDAP-QA-0626

Applicant Ref: NONE

Learning Objective: N/A

Question source: New

Question History: Not used on 2011 or 2008 SQ
Written

Cognitive level: Memory/Fundamental knowledge
Comprehension/Analysis: X

10CFR 41.12 / 43.4 / 45.10

Comments:

Question 69

Operator A has just removed a Clearance with the help of Operator B as a peer checker. The Clearance Writer is ready to have the Clearance removal Independently Verified (IV).

To independently verify the Clearance removal, the Clearance Writer may use:

- A. Either Operator A or Operator B IF the verification is performed on a different day
- B. Operator B, IF Operator A is not involved in the verification process
- C. Operator B, IF the verification is performed on a different day
- D. Neither Operator A nor Operator B

Answer:

A is incorrect. Regardless of separation of time, both Operator A and Operator B were either the performer or in close proximity to the performer, contrary to NDAP-QA-0027

B is incorrect. Operator B was in close proximity at the time the performer acted, not eligible to be independent verifier per NDAP-QA-0027

C is incorrect. Regardless of separation of time, Operator B was in close proximity at the time the performer acted, not eligible to be independent verifier per NDAP-QA-0027

D is correct. Neither operator is eligible to be independent verifier because both operators were either the performer or in close proximity at the time the performer acted per NDAP-QA-0027

From NDAP-QA-0027:

At Risk Practices to Avoid

- 2.1 Independent Verifier is in close proximity at the time the performer acts.
- 2.2 Performer and Independent Verifier walk to the component location together before the initial act.
- 2.3 Performer informs the Independent Verifier of what has or has not been done before the IV.
- 2.4 Performer is less attentive to the action, believing the Independent Verifier will catch any problems.
- 2.5 Performer leaving clues such as open panel doors, highlighted drawings, flagging, etc. which may bias the independent verifier.

K/A: 2.2.13 4.1/4.3

K/A Statement: Knowledge of clearance and tagging procedures

References: NDAP-QA-0322 Rev. 35
NDAP-QA-0027 Rev. 10

Applicant Ref: None

Learning Objective: None

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehension/Analysis:

10CFR55 41.10/45.13

Comments:

Question 70

An operator is needed to perform a filter alignment in an area where the general area radiation level is 60 mR/hr.

- The job will take 30 minutes for one operator, but can be completed in 20 minutes with two operators.
- Area radiation levels could be reduced to 30 mR/hr with shielding.
- It will take 20 minutes for one individual to install shielding.

To ensure the job dose is maintained “As Low As Reasonably Achievable”, ____ (1) ____ operator(s) should perform the lineup ____ (2) ____ shielding.

Consider total personnel dose only.

- A. (1) one
(2) with
- B. (1) one
(2) without
- C. (1) two
(2) with
- D. (1) two
(2) without

K&A Rating: 2.3.12 (3.2/3.7)

K&A Statement: **Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc..**

Justification:

- A. **Incorrect but plausible:** 1 individual to install shielding is 20 mR/hr (20 minutes in a 60 mR/hr field), plus one individual to perform lineup is 15 mR/hr (30 minutes in a 30 mR/hr field), for a total exposure of 35 mR/hr.
- B. **Correct:** 1 individual to perform lineup without shielding is a total exposure of 30 mR/hr (30 minutes in a 60 mR/hr field).
- C. **Incorrect but plausible:** 1 individual to install shielding is 20 mR/hr (20 minutes in a 60 mR/hr field), plus two individuals to perform lineup is 20 mR/hr (20 minutes in a 30 mR/hr field, times 2), for a total exposure of 40 mR/hr.
- D. **Incorrect but plausible:** 2 individuals to perform lineup without shielding is a total exposure of 40 mR/hr (20 minutes in a 60 mR/hr field, times 2)

References: N/A

Applicant Ref: NONE

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X

10CFR 41.12 / 45.9 / 45.10

Comments:

Question 71

Which one of the following describes the indicating light(s) and its (their) operation for an Area Radiation Monitor (ARM) Indicator and Trip Unit?

- A. There is a single amber alarm light which illuminates when the setpoint is reached, or loss of power occurs. The alarm light is reset when the operator resets the ARM Indicator and Trip Unit alarm
- B. There is a single amber alarm light which illuminates when the setpoint is reached, or a loss of power occurs. The light automatically resets when the initiating condition clears
- C. There are two lights; an amber light that illuminates when the setpoint is reached or a loss of power occurs, and a white light that illuminates if the detector is downscale. The alarm light is reset when the operator resets the ARM Indicator and Trip Unit alarm
- D. There are two lights; an amber light that illuminates when the setpoint is reached or a loss of power occurs, and a white light that illuminates if the detector is downscale. The lights automatically reset when the initiating condition clears

Answer:

A is incorrect. Misconception, only the Local Alarm and Auxiliary Unit has a single amber light.

B is incorrect. Misconception, only the Local Alarm and Auxiliary Unit has a single amber light and alarm must be manually reset at the Indicator and Trip Unit

C is correct. The Indicator and Trip Unit contains both an amber alarm light and a white downscale light. Alarm must be manually reset at the Indicator and Trip Unit

D is incorrect. While the Indicator and Trip Unit contains both an amber alarm light and a white downscale light, the alarm does not automatically reset and must be manually reset at the Indicator and Trip Unit

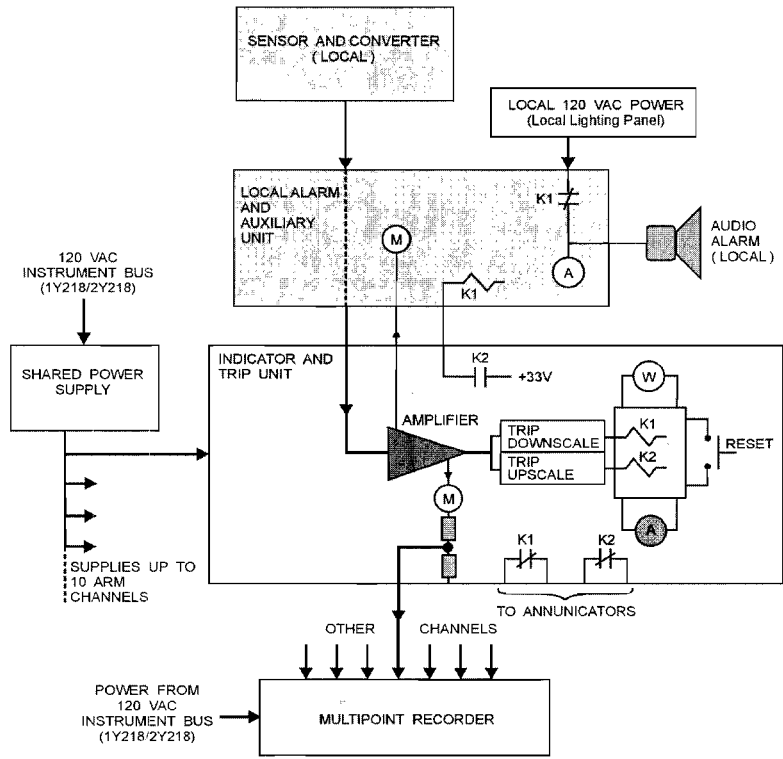
LOCAL ALARM AND AUXILIARY UNITS

All ARM Channels, with the exception of the Post-Accident ARM Channels, have Local Alarm and Auxiliary Units. Each unit is located near its associated channel Sensor and Converter Unit. Each contains a meter, an amber indicating light, and alarm circuitry. The upscale trip signal from the Indicator and Trip Unit actuates an internal relay, which initiates the audible alarm, and lights the trip indicator lamp to warn personnel that a high radiation condition exists. Local radiation level is indicated on the Auxiliary Unit Meter. The amber alarm light and horn are powered from a local 120 VAC lighting panel power source. All other power is obtained from the Indicator and Trip Unit.

INDICATOR AND TRIP UNITS

The Indicator and Trip Unit receives a DC signal from the Sensor and Converter Unit corresponding to the intensity of the gamma radiation. Each indicator and trip unit displays the sensed radiation level on a front panel meter, and with the exception of the TSC, Post-Accident and Condensate Filtration ARMs, provide an output to a Multipoint Recorder. All channels, with the exception of the Post-Accident ARMs, also supply an output to an associated auxiliary unit. Trip circuits actuate local and remote alarms in the event that an abnormal radiation level is detected, or the detector fails downscale.

Each Indicator and Trip Unit has two lights to indicate an alarm condition. The amber light is for the upscale (high) alarm, and the white light is the downscale (low) alarm. The alarm lights seal in, and must be reset manually when the alarming condition clears. Annunciator alarms are received in the Control Room for either an Upscale or a Downscale condition.



----- INDICATES SIGNAL PASSES THROUGH WITHOUT PROCESSING

(M) = METER

* Not representative of Post Accident or Condensate Filtration channels.

K/A: 2.3.5 3.9/2.9

K/A Statement: Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

References: TM-OP-079B-ST Rev. 2 Applicant Ref: None

Learning Objective: TM-OP-079B-OB Rev. 1: 10407

Question source: Modified

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehension/Analysis:

10CFR55 41.11/41.12/43.4/45.9

Comments:

Modified SSES bank question 2527

Question 72

Unit 1 is operating at 100% power. Spent fuel pool cask loading evolution is ongoing. An operator reports out a cask drop accident. The following reactor building area radiation monitors are in high alarm:

- Cask storage area (Hi Alarm) radiation reading 10 R/Hr
- Fuel pool area (Hi Alarm) radiation reading 10 R/Hr
- All other reactor building radiation levels are below alarm setpoint.

Which EOP(s) must be entered AND which one of the actions is required as a DIRECT result of the radiation levels?

- A.
 - Enter EO-100-104, "Secondary Containment Control"
 - Shutdown the Reactor IAW GO-100-004
- B.
 - Enter EO-100-102, "RPV Control"
 - Shutdown the Reactor IAW GO-100-004
- C.
 - Enter EO-100-104, "Secondary Containment Control", AND EO-100-102, "RPV Control"
 - Scram Reactor
- D.
 - Enter EO-100-104, "Secondary Containment Control" ONLY
 - Scram Reactor

K&A Rating: 2.4.2 (4.6/4.8)

K&A Statement: **Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.**

Justification:

- A. **Correct:** Secondary containment control is entered when any RB area radiation levels goes above hi alarm and reaching max safe level. Since RB area rad exceeds max level for 2 areas and no primary system is discharging into the RB, secondary containment control EOP warrants shutdown of the reactor IAW GO-100-104.
- B. **Incorrect but plausible:** Plausible if the applicant does not recall the criteria for entry in to secondary containment control and does not recall that it would take primary system discharge into RB to enter scram actions.
- C. **Incorrect but plausible:** Plausible if the applicant does not recall the criteria for entry in to secondary containment control and does not recall that it would take primary system discharge into RB to enter scram actions
- D. **Incorrect but plausible:** Plausible if the applicant does not recall the criteria for entry in to secondary containment control and does not recall that it would take primary system discharge into RB to enter scram actions

References: EO-100-104

Applicant Ref: NONE

Learning Objective: 14583, EOP entry conditions

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X

10CFR 41.7 / 45.7 / 45.8

Comments:

Question 73

Unit 1 is operating at 80% power when the following VALID Main Control Room annunciators alarm:

- AR-101-001 (A06) CONDENSATE PUMP A TRIP
- AR-101-001 (A17) RX WATER HI LEVEL
- AR-106-001 (B04) STATOR COOLING WATER OUTLET HEADER HI TEMP

Which of the following describes the appropriate operator action?

- A. Reduce reactor power by 10% RTP IAW RE instructions in CRC book
- B. Verify A and B Recirc Pumps run back to 48%
- C. Place the Mode Switch in Shutdown
- D. Take manual control of feedwater to restore reactor water level back to 35"

Answer:

A is incorrect. Without the other listed indications, this answer would be correct for AR106-001 (B04) Stator Cooling Water Outlet Header HI Temp. This alarm response requires lowering reactor power by 10% RTP IAW RE instructions in the CRC book. The listed conditions will cause a reactor scram, not allowing for a 10% reduction in power.

B is incorrect. Without the other listed indications, this answer would normally be correct for AR-101-001 (A06) CONDENSATE PUMP A TRIP. A condensate pump trip will cause a 48% Limiter 2 Recirc runback. The listed conditions will cause a reactor scram, and a 30% Limiter 1 Recirc runback on either low reactor water level (13") or low total feedwater flow (<16.4% for >20 seconds).

C is correct. The above listed indications will cause a reactor scram due to reactor feed pump and main turbine trip on high reactor water level (54"), which will then cause a reactor scram due to turbine trip with reactor >26% power.

D is incorrect. Normally per ON-145-001, if the Feedwater Master Controller fails in AUTO, the operator is directed to take manual control to restore normal feedwater level. In this case, the reactor has scrammed, and the operator will instead be taking level control IAW ON-100-101 in a startup level control alignment and maintaining level between +13" and +54". This is also incorrect because all RPFT are currently tripped on high reactor water level and must be restored before level can be recovered using Feedwater

K/A: 2.4.45

4.1/4.3

K/A Statement: Ability to prioritize and interpret the significance of each annunciator or alarm

References: ON-100-101 Rev. 26
AR-101-001 Rev. 43
AR-106-001 Rev. 47
AR-102-001 Rev. 30

Applicant Ref: None

Learning Objective: TM-OP-045I-OB Rev. 1: 16000i

Question source: Bank PB 2/07 Question 75

Question History: None

Cognitive level: Memory/Fundamental knowledge:

Comprehension/Analysis: X

10CFR55 41.10/43.5/45.3/45.12

Comments:

PB 2/07 question 75

Question 74

With respect to normal operation of the Computer UPS Static Transfer Switch:

A loss of 250 VDC Bus 1D652 results in distribution panel 1Y619 _____(1)
and upon regaining 1D652 power (with CB-1 main power supply breaker closed)
(2)_____.

- A. (1) automatically transferring to alternate supply
(2) automatically transferring back to Bus 1D652
- B. (1) requiring manual transfer to alternate supply 480 VAC MCC 1B236
(2) automatically transfers back to Bus 1D652
- C. (1) automatically transferring to alternate supply 480 VAC MCC 1B236
(2) requires manual transfer back to Bus 1D652
- D. (1) requiring manual transfer to alternate supply 480 VAC MCC 1B236
(2) requires manual transfer back to Bus 1D652

K&A Rating: 262002 K6.03 (2.7/2.9)

K&A Statement: **Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.):** D.C. electrical power

Justification:

- A. **Correct:** A loss of 250 VDC Bus 1D652 will result in panel 1Y619 to automatically transfer to alternate supply Class 1E 480 VAC source (MCC 1B236) due to the static transfer switch automatic operation. When the preferred power supply to the inverter is restored and CB-1 is reclosed, the static transfer switch automatically transfers from the alternate source to the normal source of Bus 1D652.
- B. **Incorrect but plausible:** Plausible if the applicant does not recall that the static transfer switch automatically transfer to alternate source to maintain an uninterruptible power supply.
- C. **Incorrect but plausible:** Plausible if the applicant does not recall that the static transfer switch automatically closes to the preferred power supply when the preferred power is restored and the CB-1 is closed.
- D. **Incorrect but plausible:** Plausible if the applicant does not recall that the static transfer switch automatically closes to the preferred power supply when the preferred power is restored and the CB-1 is closed

References: TM-OP-017-ST, Rev. 02

Applicant Ref: NONE

Learning Objective: 10175, TM-OP-017-OB, LO Systems, 208/120 VAC Electrical Distribution

Question source: SQ Bank

Question History: None

Cognitive level: Memory/Fundamental knowledge: X

10CFR 41.7 / 45.7

Comments:

Question 75

At normal plant operating conditions, the following Recirc Pump seal indications are noted:

- Seal #1 pressure is 1000 psig
- Seal #2 pressure is 825 psig
- SEAL STAGE HI/LO FLOW annunciator alarming
- SEAL LEAKAGE HI FLOW annunciator is **NOT** alarming

These conditions are indicative of:

- A. First-stage seal orifice plugged
- B. Second-stage seal orifice plugged
- C. Seal #1 failure
- D. Seal #2 failure

Answer:

A is incorrect. If the first-stage seal orifice were plugged, the SEAL STAGE HI/LOW FLOW annunciator will alarm on low flow. Seal #2 seal cavity pressure will lower.

B is correct. If the second-stage seal orifice were plugged, the SEAL STAGE HI/LOW FLOW annunciator will alarm on low flow. Seal #2 cavity pressure will rise.

C is incorrect. If seal #1 failed, seal #2 pressure would approach seal #1 pressure. Seal staging flow would rise to approximately 1.1 gpm. SEAL STAGING FLOW HI/LOW FLOW annunciator will alarm at 0.9 gpm.

D is incorrect. If seal #2 failed, seal #2 pressure would drop. Seal leakage would rise causing the SEAL LEAKAGE FLOW HI FLOW annunciator to alarm at 0.1 gpm.

K/A: 2.4.31 4.2/4.1

K/A Statement: Knowledge of annunciator alarms, indications, or response procedures

References: TM-OP-064C Rev. 10

Applicant Ref: None

Learning Objective: TM-OP-064C-OB Rev. 2: 2511/2522r

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:

 Comprehension/Analysis: X

10CFR55 41.10/45.3

Comments:

Question 76

Unit 1 is at 80 % power with recirculation flow being controlled with the individual controllers.

- "A" RRP speed is 66 percent
- "B" RRP speed is 72 percent.

The following transient occurs due to human performance error caused by an I&C technician.

- ICS processor pair failure occurs for "B" RRP
- Scoop tube for Recirc MG Set B locks up
- "A" RRP runback to 48 percent occurs
- Loop "A" total Jet Pump flow is 15 million lbm/hr.
- Loop "B" total Jet Pump flow is 55 million lbm/hr.
- Total indicated core flow 30 million lbm/hr
- Core Plate DP on NJP51 is 2.0 psi.

Two hours have been elapsed since the transient, MCPR is 1.11. Which of the following is the correct TS required action(s)?

- A. Be in MODE 3 within 12 hours
- B. Immediately place reactor mode switch in the shutdown position
- C. Restore compliance with MCPR AND insert all insertable control rods
- D. Restore "A" loop total Jet Pump flow to 50 million lbm/hr

K&A Rating: 295001 AA1.01 (3.5/3.6)

K&A Statement: **Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Recirculation system**

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not recall the recirculation loop jet pump flow mismatch requirements. Applicant may determine that the flow mismatch requirements have not been met, and since 2 hours has elapsed, therefore no single loop recirculation loop required limits and setpoints established within required time, required action is to be in MODE 3 within 12 hours.
- B. **Incorrect but plausible:** Plausible if the applicant does not recall the recirculation loop jet pump flow mismatch requirements identifying one recirculation loop inoperable. When one recirculation loop limits applied with safety limits, a MCPR safety limit has been violated. Safety limit violation requires inserting all insertable control rods AND restoring compliance.
- C. **Correct:** Recirculation loop jet pump flow mismatch requirements require one recirculation loop to be inoperable. When one recirculation loop limits applied with safety limits, a MCPR safety limit has been violated. Safety limit violation requires inserting all insertable control rods AND restoring compliance.
- D. **Incorrect but plausible:** Plausible if the applicant does recall the recirculation loop jet pump flow mismatch requirements and determines that compliance with flow requirements needs to be met, however, SL violation requires inserting all insertable control rods and restoring MCPR compliance.

SRO Only Justification: This question is SRO only as it requires application of Required Actions and Surveillance Requirements in accordance with rules of application requirements. This question cannot be answered solely by knowing 2 hours TS/TRM actions, LCO/TRM information listed "above the line", or TS Safety Limits.

References: TS 3.4.1

Applicant Ref: TS
section 3.4.1 (LCO and
Action statements
ONLY) NO SR or SL
section

Learning Objective:

Question source: New.

Question History: Not used on 2011 or 2008 SQ
Written

Cognitive level: Memory/Fundamental knowledge
Comprehension/Analysis: X

10CFR 41.7, 45.6

Comments:

Question 77

Unit 1 and 2 are at rated power. 'B' EDG is out of service for overhaul and 'E' EDG has been lined up to substitute. The following conditions for DG 'E' Battery Bank 0D595 are reported to the control room:

- Float voltage of one connected cell is 2.05 VDC
- Float voltage of one pilot cell is 2.09 VDC
- Float current is 2.2A
- All other cells tested were within allowable values

Assuming that the above battery conditions can not be corrected, which one of the following describes the latest that the plant can be in mode 3 and still meet technical specifications?

- A. 84 hours
- B. 86 hours
- C. 98 hours
- D. 110 hours

Answer:

A is incorrect. If the applicant immediately declares the EDG INOPERABLE, required time to reach mode 3 would be 72 hours to restore the EDG to operable per TS 3.8.1.B plus 12 hours to be in mode 3 per TS 3.8.1.F

B is correct. The applicant should initially enter TS 3.8.6, conditions A, B, and F based upon the initial conditions. Condition F is more restrictive and encompasses both initial conditions of low battery voltage and high float current. This requires declaring the battery inoperable immediately vice having time to correct the issues in Conditions A and B. The battery is immediately declared INOPERABLE per F.1. Per TS 3.8.4 Condition F, once the battery is declared INOPERABLE, 2 hours is allowed to declare the EDG INOPERABLE. Once the EDG is declared INOPERABLE in TS 3.8.1 Condition B, 72 hours is allowed to restore the EDG to OPERABLE status, or be in Mode 3 in 12 hours. $2+72+12=86$ hours

C is incorrect. Incorrect application of TS 3.8.6 condition B will result in this answer being chosen. Condition B requires these actions for float current >2 amps. Condition F correctly encompasses the concurrent conditions of one or more battery cells float voltage <2.07VDC and float current >2 amps. Allowed completion time is 12 hours for TS 3.8.6 Condition B, 2 hours to declare EDG INOPERABLE per TS 3.8.4 Condition F, 72 hours for EDG once declared INOPERABLE per TS 3.8.1 Condition B, and 12 hours to enter Mode 3 per TS 3.8.1 Condition F. $12+2+72+12=98$ hours

D is incorrect. Incorrect application of TS 3.8.6 condition A will result in this answer being chosen. Condition A requires these actions for cell voltage < 2.07 VDC. Condition F correctly encompasses the concurrent conditions of one or more battery cells float voltage <2.07 VDC and float current >2 amps. Allowed completion time is 24 hours for TS 3.8.6 Condition A, 2 hours to declare the EDG INOPERABLE per TS 3.8.4 Condition F, 72 hours for EDG once declared INOPERABLE per TS 3.8.1 Condition B, and 12 hours to reach Mode 3 per TS 3.8.1 Condition F. $24+2+72+12=110$ hours.

K/A: 295004 AA2.02 Partial or Complete Loss of D.C. Power 2.8/2.9

K/A Statement: Ability to determine and/or interpret the following as they apply to Partial or Complete Loss of D.C. Power: Extent of partial or complete loss of D.C. power

SRO Only Justification: This question is SRO only as it requires application of Required Actions and Surveillance Requirements in accordance with rules of application requirements. This question cannot be answered solely by knowing ≤ 1 hour TS/TRM actions, LCO/TRM information listed "above the line", or TS Safety Limits.

References: TM-OP-002-ST Rev. 5 Applicant Ref: TS 3.8.1
 TS 3.8.1 TS 3.8.4
 TS 3.8.4 TS 3.8.6
 TS 3.8.6

Learning Objective: TM-OP-002-OB Rev. 1: 13017

Question source: New

Question History: Not used on 2011 or 2008 SQ
Written

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.10/43.5/45.13

Comments:

See SSES Bank question 1414

Question 78

Unit 1 is operating at 100% power when the following condition occurs:

- The 1A TBCCW (P103A) tripped due to a motor fault
- The 1B TBCCW (P103B) could NOT be started
- Instrument Air System pressure is 85 psig
- Condensate Pump A (1P102A) bearing temperature is 270 F
- Condensate Pump B (1P102B) bearing temperature is 240 F
- Condensate Pump C (1P102C) bearing temperature is 230 F
- Condensate Pump D (1P102D) bearing temperature is 250 F
- EHC Fluid Temp TI-10180 is 160 F

Which of the following actions are required by ON-115-001, "Loss of TBCCW"?

- A. Commence Plant Shutdown in accordance with GO-100-104.
- B. Perform ON-100-101, "Scram Scram Imminent" AND Shutdown Condensate System IAW OP-144-001, "Condensate system".
- C. Perform ON-100-101, "Scram Scram Imminent".
- D. Reduce Reactor Power IAW Reactor Engineering instructions in CRC book and remove Condensate Pump A IAW OPO-144-001.

K&A Rating: 295018 AA2.02 (3.1/3.2)

K&A Statement: **Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER** : Cooling water temperature

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not assess the plant condition correctly to determine that one of the condensate pump bearing temperature exceeds ON action requirements to initiate a power reduction IAW CRC book. However, Commencing plant shutdown would be required if the EHC temp reached 175 F.
- B. **Incorrect but plausible:** Plausible if the applicant does not assess the plant condition correctly to determine that ONLY one of the condensate pumps bearing temperature exceeds ON action requirements. If both of the condensate pumps bearing temperatures were to exceed 265 F, then the answer would be correct to Scram and shutdown condensate system.
- C. **Incorrect but plausible:** Plausible if the applicant does not assess the plant condition correctly to determine that one of the condensate pump bearing temperature exceeds ON action requirements to initiate a power reduction IAW CRC book. However, performing ON-100-101, "Scram Scram Imminent" would be required if instrument system air pressure decrease below 65 psig.
- D. **Correct:** If bearing temperature reaches 265 F on ONE condensate pump ONLY, then ON-115-001 requires operators to reduce Rx power IAW reactor engineering instructions in CRC book.

SRO Only Justification: This question is SRO only since SRO candidates must assess plant conditions and select the correct procedure or section of a procedure to mitigate or recover the event. This question cannot be answered solely by knowing system knowledge, AOP/EOP entry conditions, or sequence of events/mitigative strategy of a procedure.

References: ON-115-001

Applicant Ref:
None

Learning Objective: 1731, TM-OP-015-OB, LO Systems, Turbine Building Closed Cooling Water

Question source: New.

Question History: Not used on 2011 or 2008 SQ Written

Cognitive level: Memory/Fundamental knowledge
Comprehension/Analysis: X

10CFR 41.10 / 43.5 / 45.13

Comments:

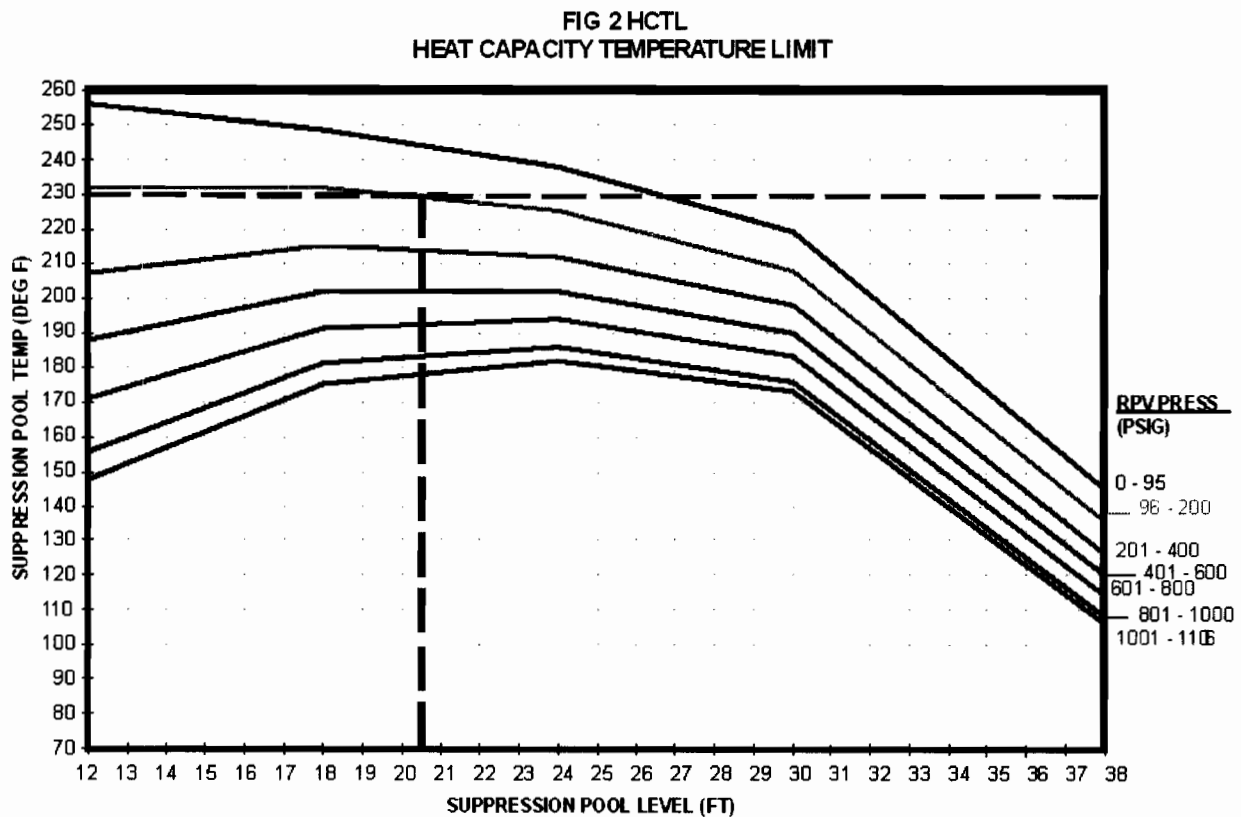
Question 79

Given the following conditions:

- Unit 1 ATWS in progress
- Initial ATWS power 4%
- RPV Pressure is 1000 psig down slowly
- Suppression Pool level is 21 feet up slowly
- Suppression Pool temperature is 180°F up slowly

Given the Heat Capacity Temperature Limit (HCTL) curve from EO-100-103, PRIMARY CONTAINMENT CONTROL, select the necessary corrective action:

- A. Perform EO-100-112, Rapid Depressurization
- B. Depressurize by opening all BPV irrespective of cooldown rate
- C. Use SRV/BPV to maintain RPV pressure below HCTL, not to exceed 100°F/hr
- D. Use SRV/BPV to maintain RPV pressure below HCTL, exceeding 100°F/hr if necessary



Question 80

Given the following conditions on Unit 1:

- A transient occurred that has caused Suppression Pool Level to lower.
- Attempts to raise Suppression Pool level in accordance with OP-159-001 Suppression Pool Cleanup System is in progress.
- RCIC was placed on min flow at 18 feet.
- HPCI is injecting and is needed to assure adequate core cooling
- At 17 feet in the Suppression Pool level, Rx SCRAM was performed IAW ON-100-101 and EO-100-102, RPV control was entered at step RC-1.
- Suppression Pool Level is now 15 feet and steady.

In accordance with EO-100-103, "Primary Containment Control", which of the following action is required and what is the basis for the required action?

Required Action	Basis
A. Rapid Depressurization	Downcomer openings are actually becoming uncovered and the ability to quench a LOCA is definitely threatened
B. Depressurize by opening all BPV irrespective of cooldown rate	Rapid Depressurization is anticipated
C. Trip RCIC Only	HPCI is needed for adequate core cooling
D. Trip HPCI Only	To prevent exhaust Sparger from uncovering which can pressurize suppression chamber and possibly cause the primary containment to fail

K&A Rating: 295030 2.4.18 (3.5/4.4)

K&A Statement: **Knowledge of the specific bases for EOPs: Low Suppression Pool Water Level**

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not assess the plant condition correctly to determine that Rapid Depressurization is required when suppression pool level reaches 12 feet.
- B. **Incorrect but plausible:** Plausible if the applicant does not assess the plant condition correctly to determine that Rapid Depressurization is not anticipated based on level stabilizing. If suppression pool level cannot be maintained above 12 inches than the anticipated Rapid Depressurization step of depressurizing with BPV is warranted.
- C. **Incorrect but plausible:** Plausible if the applicant does not assess the plant condition correctly to determine that RCIC is not required to be tripped because the exhaust flowrate of RCIC is approximately equal to that of decay heat, and is thus within the primary containment vent capability.
- D. **Correct:** Operation of HPCI with its exhaust Sparger uncovered (17 ft) will cause the suppression chamber pressure to increase and possibly cause the primary containment to fail. Therefore, HPCI must be secured irrespective of adequate core cooling.

SRO Only Justification: This question is SRO only since SRO candidates must assess plant conditions and select the correct procedure or section of a procedure to mitigate or recover the event. This question cannot be answered solely by knowing system knowledge, AOP/EOP entry conditions, or sequence of events/mitigative strategy of a procedure.

References: EO-100-103, Rev. 13
EO-000-103, Rev. 8

Applicant Ref:
None

Learning
Objective:

Question source: New.

Question History: Not used on 2011 or 2008 SQ Written

Cognitive level: Memory/Fundamental knowledge
Comprehension/Analysis: X

10CFR 41.10 / 43.1 / 45.13

Comments:

Question 81

During an ATWS, when is it permitted to use Core Spray for injection to the RPV?

- A. Restoring level between +13" and +54" following injection of Hot Shutdown Boron Weight if level was maintained between -60" and -161"
- B. When RPV level cannot be maintained >-161" using Table 15 Systems
- C. When maintaining between -60" and -110" and initial ATWS power <5 percent
- D. Core Spray injection into the RPV is not permissible during an ATWS event

Answer:

A is incorrect. Per LQ/L-16, when Hot Shutdown Boron Weight is injected, level may be restored and maintained between +13" and +54", using Table 15 systems (SLC, FW, Condensate, CRD, RCIC, HPCI, LPCI). Core spray is not permitted at this time due to injection of cold, unborated water inside the core shroud.

B is correct. Per LQ/L-21, if level cannot be restored and maintained >-161" after Rapid Depress and use of Table 15 systems, additional systems are permitted for injection irrespective of vortex limits to maintain adequate core cooling.

C is incorrect. Although ATWS power <5 percent, injection of Core Spray is not permissible unless level cannot be restored and maintained >-161" after Rapid Depress and use of Table 15 systems. At this point, additional systems are permitted for injection irrespective of vortex limits to maintain adequate core cooling.

D is incorrect. Per LQ/L-21, if level cannot be restored and maintained >-161" after Rapid Depress and use of Table 15 systems, additional systems are permitted for injection irrespective of vortex limits to maintain adequate core cooling.

K/A: 295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown A2.02 4.1/4.2

K/A Statement: Ability to determine and/or interpret reactor water level as it applies to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown

SRO Only Justification: This question is SRO only since SRO candidates must assess plant conditions and select the correct procedure or section of a procedure to mitigate or recover the event. This question cannot be answered solely by knowing systems knowledge, AOP/EOP entry conditions, or sequence of events/mitigative strategy of a procedure

References: EO-100-113 Rev. 10
EO-000-113 Bases Rev. 8

Applicant Ref: None

Learning Objective: PP002: 14622

Question source: SSES bank #1010

Question History: Not used on SSES 2008 or
2011 written exams

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.10/43.5/45.13

Comments:

Question 82

At 1530 an unisolable steam leak occurred in the Turbine Building. The blowout panels opened and OSCAR is evaluating the release. No confirmation of fuel damage. At 1550 the combined Noble Gas SPING data is calculated to be $2.3 \text{ E}8 \text{ } \mu\text{Ci}/\text{min}$.

Reactor Conditions are as follows:

- All Rods in.
- "A" MSL CANNOT be isolated.
- RPV Pressure is 800 psig lowering slowly.
- RPV Level is 40 inches and steady.
- RCIC is injecting.

Which one of the following is the correct course of action while waiting for Offsite Dose Calculations?

- A. Open ADS valves to Rapidly Depressurize
- B. Use SRVs to Cooldown $< 100^\circ\text{F} / \text{hr}$
- C. Maintain RPV pressure between 800 to 1,000 psig
- D. Open BPVs irrespective of cooldown rate

K&A Rating: 295038 2.1.20 (4.6/4.6)

K&A Statement: **Ability to interpret and execute procedure steps:** High Off-Site Release Rate

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not assess the plant condition correctly to determine that per EO-100-105, Radioactivity Release Control, table noble gas level is above Alert rate and entry into EO-100-102 is directed. Rapid depressurization is directed when noble gas levels are above $6.2 \text{ E9 } \mu\text{Ci}/\text{min}$
- B. **Correct:** EO-100-105, Radioactivity Release Control, table noble gas level is above Alert rate but below GE levels. Entry into EO-100-102 is directed, which directs to use SRVs to cooldown $< 100^\circ\text{F}/\text{hr}$. No confirmation of fuel damage is reported, which would not require entry in to EO-100-112, Rapid depressurization.
- C. **Incorrect but plausible:** Plausible if the applicant does not assess the plant condition correctly to determine that per EO-100-105, Radioactivity Release Control, table noble gas level is above Alert rate and entry into EO-100-102 is directed, which directs operators to depressurize $< 100^\circ\text{F}/\text{hr}$ using SRVs. EO-000-102 directs operator, if a primary system is discharging into an area outside of primary or secondary containments, to depressurize and initiate RPV cooldown to reduce the driving force and radioactivity of the discharge through the break in the primary system (Referece: PSTG RR-1 and EO-000-105, RR-5). RPV control does not direct operator to maintain pressure between 800 to 1,000 psig.
- D. **Incorrect but plausible:** Plausible if the applicant does not assess the plant condition correctly to determine that per EO-100-105, Radioactivity Release Control, table noble gas level is above Alert rate and entry into EO-100-102 is directed. Rapid depressurization is directed when noble gas levels are above $6.2 \text{ E9 } \mu\text{Ci}/\text{min}$, and anticipated Rapid Depressurization actions to open BPVs is performed prior to reaching GE criteria.

SRO Only Justification: This question is SRO only since SRO candidates must assess plant conditions and select the correct procedure or section of a procedure to mitigate or recover the event. This question cannot be answered solely by knowing system knowledge, AOP/EOP entry conditions, or sequence of events/mitigative strategy of a procedure.

References: EO-100-105, Rev. 4
EO-100-102, Rev. 7

Applicant Ref: None

Learning
Objective:

Question source: SQ Bank Modified.

Question History: Not used on 2011 or 2008 SQ
Written

Cognitive level: Memory/Fundamental
knowledge
Comprehension/Analysis: X

10CFR 41.10 / 43.5 / 45.12

Comments:

Question 83

Units 1 and 2 are operating at 100%. Unit 1 has initiated a Technical Specification required shutdown due to HPCI inoperability exceeding 14 days

Given the following:

- A spurious MSIV isolation occurs
- Due to improper maintenance, MSIV closure time is much faster than design
- Peak reactor vessel pressure is 1337 psig
- Due to the pressure surge, a small drywell steam leak occurs
- Peak drywell pressure is 2.48 psig
- All actuations and isolations performed as designed

Which one of the following describes the most restrictive time requirement for notifying the NRC of this event?

- A. 15 minutes
- B. 1 hour
- C. 4 hours
- D. 8 hours

Answer:

A is incorrect. There are no 15 minute reporting requirements within NDAP-QA-0720. This is a plausible distractor for those candidates that recognize the Emergency classification condition and then select the reporting requirement to the State and Count per the Emergency Plan and not the Regulatory Reporting requirement asked for in the stem.

B is correct. The declaration of any of the emergency classes is reportable within 1 hour as described in NDAP-QA-0720. While it is not important in this case to determine which specific classification is to be made, applicant should recognize that drywell pressure >1.72 psig is cause for entry into EAL and declaration of an emergency (in this case, an alert, FA1). This is the most limiting report required to be made for this event. The injection of ECCS, Tech Spec required shutdown, and RPS actuation with the reactor critical are all 4 hour notifications IAW NDAP-QA-0720.

C is incorrect. The injection of ECCS into the reactor, RPS actuation with the reactor critical, and violation of a safety limit (>1325 psig reactor steam dome pressure) require 4 hour notifications IAW NDAP-QA-0720. This is not the most limiting report required to be made. This is a plausible distractor for those candidates that do not recognize the Emergency Classification that must be made with this event.

D is incorrect. The automatic isolations that occurred at +13", -38", and -129" require 8 hour notifications IAW NDAP-QA-0720.

K/A: 295010 High Drywell Pressure G 2.4.30 2.7/4.1

K/A Statement: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

SRO Only Justification: This question is SRO only since SRO candidates must assess plant conditions and select the correct procedure or section of a procedure to mitigate or recover the event. This question cannot be answered solely by knowing systems knowledge, AOP/EOP entry conditions, or sequence of events/mitigative strategy of a procedure

References: NDAP-QA-0720 Rev. 17

Applicant Ref: NDAP-
QA-0720

Learning Objective:

Question source: New

Question History: Not used on SSES 2008 or
2011 written exams

Cognitive level: Memory/Fundamental knowledge:
Comprehension/Analysis: X

10CFR55 41.10/43.5/45.11

Comments:

Question 84

During a plant transient, the PCO observes actual RPV Level lower to -40 inches, level has turned and is now + 10 inches and rising.

Assuming all actuation and isolations occurred as designed, which of the following actions, are required to be taken?

1. Obtain in-line conductivity measurement or grab sample once per 4 hours. (TRM 3.4.1J)
2. Analyze grab samples of primary containment atmosphere once per 12 hours AND restore required primary containment atmospheric monitoring system to OPERABLE status within 30 days. (TS 3.4.6B)
3. Restore the Containment H2 and O2 Analyzers required channel(s) to OPERABLE status within 30 days (TRM 3.3.4C)
4. Restore secondary containment to OPERABLE status within 4 hours (TS 3.6.4.1A)

- A. 1, 2
- B. 3, 4
- C. 1, 2, 3
- D. 2, 3, 4

K&A Rating: 295009 AA2.01 (4.2)

K&A Statement: **Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL:** Reactor water level

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not assess the plant condition correctly to determine that containment isolation occurs at RPV level reaching -38. Both 1 and 2 are correct, conductivity recorder is inoperable when containment isolation occurs. Containment gas analyzers and drywell floor drain sump monitoring system isolation requires primary containment atmospheric monitoring system to be inoperable. However, Containment H2 and O2 analyzers are also isolated requiring entry in to TRM 3.3.4C.
- B. **Incorrect but plausible:** Plausible if the applicant does not assess the plant condition correctly to determine that containment isolation occurs at RPV level reaching -38. Secondary containment remains operable due to containment isolation starting SBT system.
- C. **Correct:** Primary containment gas analyzers isolate requiring primary containment atmospheric monitoring system to be inoperable and conductivity recorder is inoperable due to the containment isolation. Also, Containment H2 and O2 analyzers are isolated during containment isolation signals.
- D. **Incorrect but plausible:** Plausible if the applicant does not assess the plant condition correctly to determine that containment isolation occurs at RPV level reaching -38. Plausible since 2 and 3 are correct, but Secondary containment remains operable due to containment isolation starting SBT system.

K&A Match Statement: This question matches the K&A because the applicant needs to interpret that a low reactor water level condition exists and associate the low reactor water level condition with containment isolation. The question is SRO only since it requires applicant to analyze TS required actions and terminology associated with containment isolation.

SRO Only Justification: This question is SRO only since SRO candidates must assess plant conditions and select the correct technical specification actions. This question requires applicant to analyze TS required actions and terminology. This question cannot be answered solely by knowing system knowledge.

References: ON-159-002, Rev. 29
TRM 3.4.1
TS 3.4.6
TRM 3.3.4
TS 3.6.4

Applicant Ref:
TRM 3.4.1
TS 3.4.6
TRM 3.3.4
TS 3.6.4

Learning
Objective:

Question source: New.

Question History:

Cognitive level: Memory/Fundamental knowledge
Comprehension/Analysis:

X

10CFR 41.10 / 43.5 / 45.13

Comments:

Question 85

A Reactor Scram with an MSIV isolation has occurred from 100% power, and RCIC is being used for level control when the following alarms and indications are received:

- All RCIC Room Riley indicators in ALARM and rising
- RCIC Inboard, Outboard, and Bypass, Steam Isolation Valves closed
- RCIC Steam Header pressure 900 psig constant
- RCIC INJECTION VALVE HV149F013, open
- RCIC Pump suction pressure upscale constant
- SIMPLEX Panel audible alarm
- AR-101-001 (B05), RX BLDG AREA PANEL 1C605 HI RADIATION
- AR-036-001 (B01), PUMP IS OPERATING
- AR-016-001 (G15), FIRE PROTECTION PANEL OC650 SYSTEM TROUBLE

What transient has occurred, and what actions are required?

- A. Feedwater is discharging at a break at RCIC suction
Start all Unit Coolers
Shut RCIC Injection Valve HV149F013
Monitor RCIC room temperatures lowering
- B. Break in steam line discharging into RCIC room
Trip RCIC
Start all Unit Coolers
Monitor RCIC room temperatures lowering
- C. Break in steam line discharging into RCIC room
Monitor SIMPLEX panel
Start all Unit Coolers
Monitor RCIC room temperatures lowering
- D. Feedwater discharging at a break at RCIC suction or fire in RCIC room
Initiate MANUAL RCIC isolation
Start all Unit Coolers
Monitor RCIC room temperatures lowering

K&A Rating: 295032 High Secondary Containment Area Temperature G2.2.44 (4.4)

K&A Statement: **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**

Justification:

- A. **Correct:** A break has occurred in the RCIC room. The steam pressure at 900 psig in the steam piping eliminates the possibility of a steam line break. A fire would not produce a high suction pressure. A RCIC suction line break with feedwater flowing back through RCIC to the suction would release hot water and steam into the RCIC room. Actions per EO-100-104 would increase room cooling, isolate the leak by closing the injection valve which must be done manually.
- B. **Incorrect but plausible:** Plausible if applicants believe that a steam line break has occurred due to high room temperatures and isolations.
- C. **Incorrect but plausible:** Plausible if applicants believe that a steam line break has occurred due to high room temperatures and isolations.
- D. **Incorrect but plausible:** Plausible if applicants believe that a manual RCIC isolation will isolate the leak. Manual RCIC isolation will close the steam valves but will not close the RCIC water flow path, which is the source of the leak.

SRO Only Justification: This question is SRO only as it cannot be answered solely by knowing ≤ 1 hour TS/TRM actions, 'above the line' LCO/TRM information, TS safety limits. This question requires knowledge of TS bases that is required to analyze TS required actions and terminology.

References: EO-100-104, Rev. 6

Applicant Ref:
None

Learning Objective: PP002: 14594/14596

Question source: Bank, SSES 2004 exam

Question History: Not used on SSES 2008 or 2011 written exams

Cognitive level: Memory/Fundamental knowledge
Comprehension/Analysis:

X

10CFR 41.5 / 43.5 / 45.12

Comments:

Question 86

Unit 1 is in Mode 2 with preparations for a Reactor Startup in progress. SO-153-004, "Quarterly SBLC Flow Verification", is being performed and the breaker for Standby Liquid Control Pump 1P208A tripped when the NPO started the pump and it will NOT reset.

Which of the following describes the PRESENT Tech Spec condition(s) and required actions?

A. LCO 3.1.7 Condition C, ONLY

Return SBLC System B to standby alignment IAW SO-153-004

B. LCO 3.1.7 Condition C, ONLY

Align alternate injection flowpath via SBLC Demin Cross tie IAW OP-153-001

C. LCO 3.1.7 Conditions B and C

Return SBLC System B to standby alignment IAW SO-153-004

D. LCO 3.1.7 Conditions B and C

Align alternate injection flowpath via SBLC Demin Cross tie IAW OP-153-001

K&A Rating: 211000 A2.03 (3.2/3.4)

K&A Statement: **Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:**
Pump trip

Justification:

- A. **Incorrect but plausible:** Plausible due to partial correct, during performance of the SO, the SLC Pump suction is isolated from the SLC Storage Tank and their discharges to the RPV are closed, therefore 3.1.7 C applies because neither pump can inject to the RPV. However, "B" also applies due to the tripped breaker. Also the required actions would be to restore SBLC system B to standby alignment IAW SO-153-004 due to B system pump suction is isolated and require realignment.
- B. **Incorrect but plausible:** Plausible due to partial correct, during performance of the SO, the SLC Pump suction is isolated from the SLC Storage Tank and their discharges to the RPV are closed, therefore 3.1.7 C applies because neither pump can inject to the RPV. However, "B" also applies due to the tripped breaker. Aligning alternate injection path is not used to restore operability for the B system, however, alternate injection line is utilized for EOPs..
- C. **Correct:** During performance of the SO, the SLC Pump suction is isolated from the SLC Storage Tank and their discharges to the RPV are closed, therefore 3.1.7 C applies because neither pump can inject to the RPV. "B" also applies due to the tripped breaker. Also the required actions would be to restore SBLC system B to standby alignment IAW SO-153-004 due to B system pump suction is isolated and require realignment.
- D. **Incorrect but plausible:** Plausible due to partial correct, during performance of the SO, the SLC Pump suction is isolated from the SLC Storage Tank and their discharges to the RPV are closed, therefore 3.1.7 C applies because neither pump can inject to the RPV. "B" also applies due to the tripped breaker. Aligning alternate injection path is not used to restore operability for the B system, however, alternate injection line is utilized for EOPs.

SRO Only Justification: This question is SRO only since SRO candidates must assess plant conditions and select the correct TS action conditions. This question cannot be answered solely by knowing system knowledge, it requires an evaluation to understand the potential impact of current plant conditions and surveillance conditions to recover the plant to a safe condition.

References: TS 3.1.7
SO-153-004

Applicant Ref:
TS 3.1.7

Learning
Objective:

Question source: SQ Bank

Question History: Not used on 2011 or 2008 SQ Written

Cognitive level: Memory/Fundamental knowledge
Comprehension/Analysis: X

10CFR 41.5 / 45.6

Comments:

Question 87

Unit 2 is experiencing an ATWS. All attempts so far to insert control rods and inject SBLC have been unsuccessful. Plant conditions as follows:

- Reactor power is 11%
- Feedwater and condensate are unavailable
- Reactor water level is being maintained -60" → -110" using HPCI/RCIC
- Suppression Pool temperature is 144°F and rising slowly

Two minutes later, the following additional plant conditions exist:

- AR-214-E01 Condensate Storage Tank LO Water Level
- AR-214-F02 HPCI Pump Suction LO Press
- AR-214-C02 HPCI Turbine Trip Solenoid Energized
- AR-214-A01 HPCI Turbine Tripped

What procedure(s) are required to restore level control?

- A. Ensure both HPCI and RCIC turbines are tripped, perform EO-200-112 Rapid Depressurization, and use low pressure ECCS for level control
- B. Recover from HPCI Turbine trip and restore HPCI to service in accordance with OP-252-001 High Pressure Coolant Injection (HPCI) System
- C. Recover from HPCI Turbine isolation and restore HPCI to service in accordance with OP-252-001 High Pressure Coolant Injection (HPCI) System
- D. Transfer HPCI suction to Suppression Pool in accordance with AR-214-E01 and restore HPCI to service in accordance with OP-252-001.

Question 88

Unit 1 is operating at 95 percent power with no current TS/TRM required action conditions.

The following conditions exist:

- A surveillance test of the reactor building recirculation system has just been performed.
- The test results were UNSATISFACTORY for the reactor building recirculation fan 0V201A pump and its associated dampers.
- A functional test of the Standby Gas Treatment (SGTS) Fire Suppression System DS-093 and DS-094 were also performed
- The rest results were UNSATISFACTORY for the SGTS Charcoal Filter 0F169B.

Which one of the following describes how these failed tests impacts the Standby Gas Treatment System and/or the plant?

- A. Both trains of SGTS remain Operable. A continuous firewatch, with backup fire suppression equipment is required to be posted for SGTS Charcoal Filter 0F169B.
- B. Both trains of SGTS remain Operable. An hourly firewatch patrol for SGTS Charcoal Filter 0F169B area is required.
- C. Both trains of SGTS are Inoperable. Restore one SGTS Train to Operable status in 4 hours, or be in Mode 3 within 12 hours.
- D. One SGT subsystem is inoperable. Restore one SGT subsystem to Operable status in 7 days, or be in Mode 3 within 12 hours.

K&A Rating: 261000 2.2.22 (4.7)

K&A Statement: **Knowledge of limiting conditions for operations and safety limits: SGTS**

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not assess the plant condition correctly. Both trains of SGTS do not remain operable due to one train of reactor building recirculation system and associated dampers inoperable. When any one reactor building recirculation division is not OPERABLE, one arbitrarily determined SGT system is not operable.
- B. **Incorrect but plausible:** Plausible if the applicant does not assess the plant condition correctly. Both trains of SGTS do not remain operable due to one train of reactor building recirculation system and associated dampers inoperable. When any one reactor building recirculation division is not OPERABLE, one arbitrarily determined SGT system is not operable.
- C. **Incorrect but plausible:** Plausible if the applicant does not assess the plant condition correctly, and determine that 'B' train of SGTS is inoperable due to its fire suppression system inoperable. Fire suppression system is not required for SGTS operability (TS B.3.6.4.3). Applicant may determine this to be plausible due to one train of reactor building recirculation system and associated dampers inoperable and 'B' train of SGT system inoperable due its suppression system failure.
- D. **Correct:** When any one reactor building recirculation division is not OPERABLE, one arbitrarily determined SGT system is not operable. However, Fire suppression system is not required for SGTS operability (TS B.3.6.4.3), so only one train of SGTS is inoperable.

SRO Only Justification: This question is SRO only since SRO candidates must assess plant conditions and select the correct TS action conditions. This question cannot be answered solely by knowing system knowledge, it requires an evaluation to understand the potential impact of current plant conditions and surveillance conditions to recover the plant to a safe condition.

References: TS 3.6.4
B.3.6.4
TRM 3.7.3.2

Applicant Ref:
TS 3.6.4

Learning
Objective:

Question source: SQ Bank Modified.

Question History: Not used on 2011 or 2008 SQ Written

Cognitive level: Memory/Fundamental knowledge
Comprehension/Analysis: X

10CFR 41.5 / 45.6

Comments:

Question 89

Technical Specifications require a minimum amount of fuel oil level for the DG A-D day tank.

Which one of the following describes the bases for the minimum required fuel oil day tank level for EDG operation?

The minimum required fuel oil day tank supply for EDG operation....

- A. Ensures adequate fuel oil for a minimum of 55 minutes of DG A-D operation at the maximum post accident load demand conditions
- B. Ensures adequate fuel oil for a minimum of 55 minutes of DG A-D operation at DG continuous rated load conditions
- C. Ensures adequate fuel oil for a minimum of 62 minutes of DG A-D operation at the maximum post accident load demand conditions
- D. Ensures adequate fuel oil for a minimum of 62 minutes of DG A-D operation at DG continuous rated load conditions

K&A Rating: 262001 AC Electrical Distribution G2.2.25 (4.2)

K&A Statement: **Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.**

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not recall that the minimum fuel oil level ensures minimum of 55 minutes of DG A-D operation at continuous rated loads, not maximum post accident loads.
- B. **Correct:** The minimum fuel oil level ensures minimum of 55 minutes of DG A-D operation at continuous rated loads.
- C. **Incorrect but plausible:** Plausible if the applicant does not recall that the minimum fuel oil level ensures minimum of 55 minutes of DG A-D operation at continuous rated loads, not maximum post accident loads, and the 62 minutes operation is for DG E.
- D. **Incorrect but plausible:** Plausible if the applicant does not recall that the minimum fuel oil level ensures minimum of 55 minutes of DG A-D operation at continuous rated loads, not maximum post accident loads, and the 62 minutes operation is for DG E.

SRO Only Justification: This question is SRO only as it cannot be answered solely by knowing ≤ 1 hour TS/TRM actions, 'above the line' LCO/TRM information, TS safety limits. This question requires knowledge of TS bases that is required to analyze TS required actions and terminology.

References: TS 3.8.1
SR 3.8.1.4
SR 3.8.1.4 Basis

Applicant Ref:
None

Learning Objective: TM-OP-024-OB Rev. 0: 12556

Question source: Bank, Nine Mile Point 8/09

Question History: Not used on SSES 2008 or 2011 written exams

Cognitive level: Memory/Fundamental knowledge X
Comprehension/Analysis:

10CFR 41.5 / 41.7 / 43.2

Comments:

Question 90

Both units are operating at full power with T-10 load tap changer in manual operation.

The following conditions occur:

- The starting air system for the DG "B" develops a leak, and the air compressor can maintain 175 psig at the receivers.
- NPO just reported that DG "E" lube oil sump level is not visible in the sight glass.

Which of the following actions is required?

- A. Enter LCO 3.0.3 immediately
- B. Restore DG B to OPERABLE status within 72 hours
- C. Restore DG B or E to OPERABLE status within 2 hours
- D. Restore Transformer T-10 or DG B to OPERABLE status within 12 hours

K&A Rating: 264000 G2.2.40 (4.7)

K&A Statement: **Ability to apply Technical Specifications for a system:** Emergency Generators

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant concludes that the both B and E DG are INOP and offsite circuit is INOP due to T-10 load tap changer manual operations.
- B. **Incorrect but plausible:** Plausible if the applicant determined that Only the "B" DG is INOP due to starting air receiver pressure falling below 180 psig. DG E is INOP due to sump level not visible, but DG E is not required DG for 3.8.1. Applicant fails to determine that for offsite circuit operability load tap changer must be in automatic operations.
- C. **Incorrect but plausible:** Plausible if the applicant concludes that B and E DG are INOP and both are required to be operable per 3.8.1.
- D. **Correct:** The "B" DG is INOP due to starting air receiver pressure falling below 180 psig and one offsite circuit is INOP due to T-10 load tap changer in manual operations.

SRO Only Justification: This question is SRO only since it applies to additional knowledge and abilities pertaining to an SRO license. Knowledge of the application of generic Limiting Condition for Operation requirements.

References: TS 3.8.1
TS 3.8.3

Applicant Ref:
3.8.1 and 3.8.3

Learning Objective:

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge X
Comprehension/Analysis:

10CFR 41.10 / 43.2 / 43.5 / 45.3

Comments:

Question 91

Unit 1 refueling operations are in progress with the following plant conditions:

- All control rods are inserted
- The Refuel Platform is over the core
- A bundle has just been grappled and lifted from the core
- Bridge movements are being controlled using the joysticks
- As the bundle clears the top guide, a malfunction of the main hoist overspeed switch causes a fault lockout
- ON-081-002, "Refueling Platform Operation Anomaly", has been entered
- The fault lockout has been reset in accordance with ON-081-002

Based on the above....

The fuel bundle _____ (1) _____. In accordance with ON-081-002, "Refueling Platform Operation Anomaly", once the fuel bundle has been placed in a safe position, permission must be granted by _____ (2) _____ to resume fuel handling activities .

- A. (1) must be placed in its previous in-core location
(2) Assistant Ops Manager – Shift with concurrence of applicable Unit Supervisor and Refuel Floor Manager
- B. (1) must be placed in its previous in-core location
(2) Shift manager with concurrence of applicable Unit Supervisor and Refuel Floor Manager
- C. (1) must be placed in the fuel pool emergency setdown location
(2) Assistant Ops Manager – Shift with concurrence of applicable Unit Supervisor and Refuel Floor Manager
- D. (1) must be placed in the fuel pool emergency setdown location
(2) Shift manager with concurrence of applicable Unit Supervisor and Refuel Floor Manager

From TM-OP-081A Fuel Handling Unit 1:

Start pushbutton is used to turn system power on. If any master joystick is out of position, i.e., a motion is commanded, the Platform will not START when the START pushbutton is depressed. Should a fault lockout condition exist, AND a master joystick is deflected, the Refueling Platform will automatically de-energize, and the START pushbutton will have to be depressed to restore power.

From ON-081-002 Refueling Platform Operation Anomaly:

Entry condition: Refuel Platform electrical failures including Fault Lockout (Both units) or Encoder error (Unit 1)

4.10 Unit 1 Refueling Platform electrical failure including Fault Lockout or Encoder error:

NOTE (1):	Following Interlocks cannot be overridden: Bridge Motor Drive Fault, Trolley Motor Drive Fault, Main Hoist Motor Drive Fault. These must be manually reset in 1C253 (See Attachment A for description of lockouts).
NOTE (2):	If a Fault Lockout exists and a joystick is defected, the Refuel Platform will shut down. The start push button must be depressed to restore power to the refuel platform.
NOTE (3):	Hoist overspeed requires manual reset at the motor.
NOTE (4):	Fault override keyswitch may be used to override any Encoder errors. Caution should be used when operating the Platform in this condition.

- 4.10.1 **Attempt** to reset Fault Lockout.
- 4.10.2 **IF** electrical failure resets:
 - a. **IF** component is above the top guide / fuel pool rack / DSC grid, **Place** component into Fuel Pool emergency setdown location: **MM-34** (Double Blade Guide location: **MM-34 & LL-33**).
 - OR**
 - b. **Place** Fuel Bundle in a location as specified by FACCTAS instruction.
 - c. **IF** component is inserted into top guide / fuel pool rack / DSC grid, **Place** component into current location.

- d. **Document** move in comments section of applicable FACCTAS step.
- e. **Notify** Reactor Engineering to evaluate impact on remaining FACCTAS steps and **Generate** FACCTAS Change Notice as appropriate.
- f. **Notify** Electrical Maintenance.
- g. **Proceed** to Section 4.15.

Once cause has been identified and corrected, ON-081-002 directs proceeding to step 4.15.

- 4.15 **Prior To** continuing further Fuel Handling Activities, **Obtain** permission from Assistant Ops manager - Shift/WCC **OR** Manager - Nuclear Operations **AND** concurrence of applicable Unit Supervisor and Refuel Floor Manager.

K&A Rating: 234000 Fuel Handling Equipment A2.03 (3.1)

K&A Statement: **Ability to (a) predict the impacts of loss of electrical power on the Fuel Handling Equipment; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations**

Justification:

- A. **Incorrect but plausible:** If candidate does not recall where fuel bundle must be returned to if it has been raised above the top guide, may choose this answer.
- B. **Incorrect but plausible:** If candidate does not recall where fuel bundle must be returned to if it has been raised above the top guide, may choose this answer. Also, Shift manager is a plausible choice for requesting permission to resume fuel handling

- activities, as the shift manager is responsible for all shift operations.
- C. **Correct:** If fuel bundle has been raised above top guide, it must be placed in the fuel pool emergency setdown position. It cannot be returned to its original core position as a precaution to prevent returning the bundle to an incorrect core location. Permission from Assistant Ops Manager – Shift and concurrence from applicable Unit Supervisor and Refuel Floor Manager is required before resuming fuel handling activities.
- D. **Incorrect but plausible:** If fuel bundle has been raised above top guide, it must be placed in the fuel pool emergency setdown position, but Shift Manager permission is insufficient to resume fuel handling activities. Plausible choice for requesting permission to resume fuel handling activities, as the shift manager is responsible for all shift operations.

SRO Only Justification: This question is SRO only as it requires knowledge of fuel handling facilities and procedures. It cannot be answered solely by knowing 'systems knowledge', immediate operator actions, AOP/EOP entry conditions, or overall sequence of events/mitigative strategy of a procedure.

References: TM-OP-081A-ST Rev. 7
ON-081-002 Rev. 19

Applicant Ref:
None

Learning

Objective: TM-OP-081A-OB Rev. 2: 12464/12465f
TM-OP-081B-OB Rev. 2: 10785/10786e

Question source: Modified VY 5/2009 exam

Question History: Not used on SSES 2008 or 2011 written exams

Cognitive level: Memory/Fundamental knowledge
Comprehension/Analysis:

X

10CFR 41.5/45.6

Comments:

Question 92

Which of the following meets the minimum requirements for Fire Suppression Water Supply Subsystem to be aligned in order for the subsystem to be considered operable?

- A. Only the Clarified Water Storage Tank (OT523) with a minimum volume of 300,000 gallons. No other source will maintain the subsystem operable.
- B. The Clarified Water Storage Tank (OT523) with a minimum volume of 300,000 gallons, AND BOTH the Unit 1 and Unit 2 Cooling Tower Basins
- C. The Clarified Water Storage Tank (OT523) with a minimum volume of 300,000 gallons, AND EITHER the Unit 1 or Unit 2 Cooling Tower Basin
- D. Only ONE of Unit 1 Cooling Tower Basin, Unit 2 Cooling Tower Basin, OR Clarified Water Storage Tank (OT523) with a minimum volume of 300,000 gallons

K&A Rating: 286000 G2.4.25 Fire Protection (3.7)

K&A Statement: **Knowledge of fire protection procedures**

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not recall that for fire suppression water operability, one water supply from either U1 or U2 cooling tower basin can be utilized besides Clarified Water Storage Tank (OT523) with a minimum volume of 300,000.
- B. **Incorrect but plausible:** Plausible if the applicant does not recall that only ONE source is required for operability, either U1 cooling tower basin, U2 cooling tower basin, OR Clarified Water Storage Tank (OT523) with a minimum volume of 300,000.
- C. **Incorrect but plausible:** Plausible if the applicant does not recall that only ONE source is required for operability, either U1 cooling tower basin, U2 cooling tower basin, OR Clarified Water Storage Tank (OT523) with a minimum volume of 300,000.
- D. **Correct:** Only ONE source is required for operability, either U1 cooling tower basin, U2 cooling tower basin, OR Clarified Water Storage Tank (OT523) with a minimum volume of 300,000.

SRO Only Justification: This question is SRO only since SRO candidates must assess plant conditions and determine the operability status of the plant equipments. This question requires knowledge of TRM basis and cannot be answered solely by knowing system knowledge, AOP/EOP entry conditions, or sequence of events/mitigative strategy of a procedure.

References: TRM 3.7.3
TRM B.3.7.3.1

Applicant Ref:
None

Learning
Objective:

Question source: SQ Bank

Question History: Not used on 2011 or 2008 SQ Written

Cognitive level: Memory/Fundamental knowledge X
Comprehension/Analysis:

10CFR 41.10 / 43.5 / 45.13

Comments:

Question 93

Unit 1 is in a refueling outage in Mode 5. There is **NO** fuel movement in progress. The 'A' CREOAS System is OPERABLE.

The 24 month surveillance for 'B' CREOAS System has just been completed at 0400 with the following test results:

- HEPA filter in place test penetration and system bypass flow is 0.03% at a flow rate of 5780 cfm
- Charcoal adsorber in place test penetration and system bypass flow is 0.05% at a flow rate of 5810 cfm

The outage schedule has an activity to perform a recirculation pump suction valve replacement using a freeze seal beginning at 0800.

You are the CRS, tasked with evaluating Tech Specs to determine if this maintenance activity can be performed as scheduled.

Which **ONE** of the following describes:

- (1) Whether this work can be authorized to commence with the existing plant conditions **AND WITHOUT** any additional evaluations, and,
- (2) The basis for that decision?

- | | | |
|----|---|---|
| A. | (1)
Maintenance activity CANNOT be authorized | (2)
LCO 3.7.3 is currently NOT met and entry into OPDRV specified condition can ONLY be done when two CREOAS subsystems are OPERABLE |
| B. | Maintenance activity can be authorized | Entry into OPDRV specified condition is allowed because TS 3.7.3 D.1 allows continued operation for an unlimited period with an OPERABLE CREOAS subsystem in pressurization/filtration mode |
| C. | Maintenance activity CANNOT be authorized | LCO 3.7.3 is currently NOT met and entry into OPDRV specified condition would require a LCO 3.0.4.b risk assessment be performed prior to performing the maintenance |
| D. | Maintenance activity can be authorized | Maintenance activity is NOT an OPDRV and Technical Specification 3.7.3 is not applicable in Mode 5 |

K&A Rating: 290003 Control Room HVAC G2.2.38 (4.6)

K&A Statement: **Knowledge of conditions and limitations in the facility license**

Justification:

- A. **Incorrect but plausible:** With one subsystem operable the work can commence without any additional evaluations. LCO 3.0.4.a allows entry into a specified condition if the actions have an unlimited completion time. With one subsystem inoperable 3.7.3.A would be the required action and if the 7 days had expired, action 3.7.3.D would allow placing the operable subsystem for an unlimited time, therefore the maintenance activity may be authorized.
- B. **Correct:** because the 'B' CREOAS is inoperable per TS 5.5.8.b and LCO 3.0.4.a is applicable. LCO 3.7.3 requires 2 subsystems to be operable. LCO 3.7.3 is not met and entry into a specified condition (during OPDRVs) can only be made IAW LCO 3.0.4. LCO 3.0.4.a allows entry into a specified condition if the actions taken have an unlimited time which per 3.7.3.D, with the operable CREOAS system in pressurization/filtration mode, OPDRVs can continue for an unlimited time. Therefore the OPDRV specified condition can be entered without restoring the inoperable subsystem or by performing a risk assessment.
- C. **Incorrect but plausible:** because with one subsystem operable the work can commence without any additional evaluations. Although the 'B' CREOAS is inoperable per TS 5.5.8.b and therefore 3.7.3 is not met, the risk assessment is not required because LCO 3.0.4.a allows entry into the specified condition. Plausible if the candidate does not consider LCO 3.0.4.a to allow entry into the specified condition of an OPDRV.
- D. **Incorrect but plausible:** because the maintenance activity is an OPDRV. Maintenance activities that create a single flow path from the RPV greater than 1" in diameter that is not isolated or isolable by plant design features and does not have an alternative physical or administrative barrier established in a PORC recommended procedure or PORC-recommended work document.

SRO Only Justification: This question is SRO only as it cannot be answered solely by knowing ≤ 1 hour TS/TRM actions, 'above the line' LCO/TRM information, TS safety limits. This question requires application of required actions and surveillance requirements in accordance with rules of application requirements.

References: TS 3.7.3
TS 5.5.7

Applicant Ref:
TS 3.7.3
TS 5.5.7

Learning Objective: TM-OP-0401-OB Rev. 2: 13426

Question source: Bank, Fitzpatrick 5/10

Question History: Not used on SSES 2008 or 2011 written exams

Cognitive level: Memory/Fundamental knowledge
Comprehension/Analysis: X

10CFR 41.7 / 41.10 / 43.1/45.13

Comments:

Question 94

Unit 1 is at 90%

- At 0800 on 2/16, it was determined that part of a TS required surveillance was NOT performed.
- The incomplete surveillance was performed on 2/13.
- The last time this surveillance was completed satisfactorily was at 0800 on 1/15.
- The surveillance is required to be performed at least once per 31 days.

The action statement requires that inoperable equipment must be restored within 72 hours, or be in Hot Shutdown within the next 12 hrs and in Cold Shutdown within the following 24 hours.

At 0800 on 2/19, the surveillance is completed, and is determined to be UNSAT.

Which of the following is required?

- A. Be in Hot Shutdown by NO later than 2000 on 2/19.
- B. Be in Cold Shutdown by NO later than 0800 on 2/20.
- C. Be in Hot Shutdown by NO later than 2000 on 2/22.
- D. Be in Cold Shutdown by NO later than 0800 on 2/23.

K&A Rating: 2.1.39 (4.3)

K&A Statement: **Knowledge of conservative decision making practices.**

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not recognize that additional 72 hours is allowed by the LCO before beginning the clock towards Hot and Cold Shutdown requirements. Applicant may determine that 72 hours starts from the time of discovery on 2/16 at 0800.
- B. **Incorrect but plausible:** Plausible if the applicant does not recognize that additional 72 hours is allowed by the LCO before beginning the clock towards Hot and Cold Shutdown requirements. Applicant may determine that 72 hours starts from the time of discovery on 2/16 at 0800.
- C. **Correct:** The surveillance was NOT completed within the required surveillance time of 31 days, so SR 3.0.3 applies, which provides additional 24 hours or surveillance specified frequency, whichever is greater. When the surveillance is performed within in the delay period and the surveillance is not met, the LCO must immediately be declared not met, and applicable conditions must be entered. Therefore, a 72 hour clock starts before the 12 hour clock begins for the requirement to be in Hot Shutdown. This would be total of 84 hours resulting in a time of 2000 on 2/22.
- D. **Incorrect but plausible:** Plausible if the applicant determines that cold shutdown is required within 24 hours from the expiration of 72 hours restoration period.

References: LCO 3.0.2
SR 3.0.3

Student Ref:

None

Learning Objective: N/A

Question source: HC 08/10 Exam Modified

Question History: HC Bank 53354

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.10 / 43.5 / 45.12

Question 95

A transient has occurred on Unit 1, which has caused major fuel damage. The following conditions exist:

- All rods are in
- MSIV's and drains are isolated
- RPV level is +29 inches and RPV pressure is 810 psig
- RCIC Equipment Area temperature is 101°F, rising slowly
- HPCI Equipment Area temperature is 103°F, rising slowly
- RCIC Equipment Area ARM is reading 13.6 R/hour, rising slowly
- HPCI Equipment Area ARM is reading 11.3 R/hour, rising slowly
- CRD North and South reading 14.2 R/hour, rising slowly

Based upon the given conditions, in accordance with EO-100-104, Secondary Containment Control:

- Is a primary system discharging into the Reactor Building?
 - What action should the Unit Supervisor direct?
- A. A primary system is discharging into the Reactor Building, cooldown the reactor in accordance with EO-100-102, RPV Control
- B. A primary system is discharging into the Reactor Building, enter and perform EO-100-112, Rapid Depressurization
- C. A primary system is NOT discharging into the Reactor Building, depressurize the reactor in accordance with EO-100-102, RPV Control
- D. A primary system is NOT discharging into the Reactor Building, cooldown the reactor in accordance with GO-100-004, Plant Shutdown to Minimum Power

K&A Rating: G 2.1.45 (4.3)

K&A Statement: **Ability to identify and interpret diverse indications to validate the response of another indication**

Justification:

- A. **Incorrect but plausible:** No indications given in the stem that indicate a primary system discharging. Plausible if applicant believes that a primary system is discharging, applicant would choose EO-100-102 if they did not recognize two or more areas above max safe, a rapid depressurization would instead be required.
- B. **Incorrect but plausible:** No indications given in the stem that indicate a primary system discharging. Plausible if applicant believes that a primary system is discharging and two or more areas above max safe radiation level, requiring rapid depressurization
- C. **Incorrect but plausible:** A primary system is NOT discharging into the Reactor Building; depressurization in accordance with EO-100-102, RPV Control is not warranted in this case. It would be required if an entry condition existed, or if a primary system was discharging concurrent with one area above max safe radiation level.
- D. **Correct:** A primary system is NOT discharging into the Reactor Building; depressurization in accordance with GO-100-104, Plant Shutdown to Minimum Power is required. Applicant may also not choose this answer due to title of GO-100-104, but is the procedurally directed procedure.

SRO Only Justification: This question is SRO only as it cannot be answered solely by knowing 'systems knowledge', immediate operator actions, AOP/EOP entry conditions, or overall sequence or mitigative strategy of a procedure. This question requires assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed.

References: EO-100-004 Rev. 6

Applicant Ref:
None

Learning Objective: PP002: 14586, 14593, 14596, 14613

Question source: Modified SSES bank 1077

Question History: Not used on SSES 2008 or 2011 written exams

Cognitive level: Memory/Fundamental knowledge
Comprehension/Analysis: X

10CFR 41.7 / 43.5/45.4

Comments:

Question 96

Both Units are operating at full power. The following sequence of events occurs:

- Unit 1 Core Spray System Loop A is tagged out of service for normal system outage window.
- The Transmission operator reports that Montour line supplying 230 kV T-10 switchyard will be removed from service within next 10 minutes for emergent work.
- Maintenance requests operations to tag out 1B120 Turbine building 480V Load Center per the online schedule

IAW NDAP-QA-1902, Maintenance Rule Risk Assessment and Management Program, what actions, if any, are required to tag out the equipment and the reason?

- A. None. A risk assessment has been performed for the work week which demonstrates that risk configuration is acceptable.
- B. Perform a new risk assessment within 12 hours of tagging out the equipment to determine any required compensatory actions.
- C. None. The 480v Non-ESS Load Center is not safety related equipment and does not require a risk assessment prior to performing maintenance.
- D. Perform a new risk assessment prior to releasing the equipment to ensure the plant risk is acceptable.

K&A Rating: 2.2.17 (3.8)

K&A Statement: **Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.**

Justification:

- A. **Incorrect but plausible:** If the applicant does not understand the overall risk assessment of the plant conditions, and determines that normal online schedule should have included overall risk assessment of the plant conditions.
- B. **Incorrect but plausible:** If the applicant determines that 480V Non- ESS load center or Montour line is not an equipment modeled in the risk assessment tools, therefore, IAW NDAP-QA-1902, a new risk assessment shall be performed within 12 hours of the emergent conditions.
- C. **Incorrect but plausible:** If the applicant determines that 480V Non- ESS load center is not a safety related equipment that is modeled in the risk assessment tools, therefore, IAW NDAP-QA-1902, a new risk assessment shall be performed within 12 hours of the emergent conditions.
- D. **Correct:** IAW NDAP-QA-1902, a new risk assessment shall be performed prior to releasing the equipment to ensure the plant risk is acceptable.

SRO Only Justification: This question is SRO only since it applies to additional knowledge and abilities pertaining to an SRO license. Knowledge of when to perform risk assessment as required by technical specifications. This question cannot be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

References: NDAP-QA-1902, Rev. 4

Applicant Ref:
None

Learning
Objective:

Question source: New.

Question History: Not used on 2011 or 2008 SQ Written

Cognitive level: Memory/Fundamental knowledge
Comprehension/Analysis: X

10CFR 41.10 / 43.5 / 45.13

Comments:

Question 97

The Unit 2 PCO is preparing a Switching Order IAW OI-AD-085. When would a 10CFR50.59 review be required?

- A. HPCI is out-of-service with a clearance applied to the steam supply and injection valves. The Switching Order will direct the NPO to open the vacuum pump supply breaker
- B. System Engineering wants to adjust the air flow on the Refuel Floor. The Switching Order will direct the NPO to throttle HD-17838A and HD-17838B by 10 degrees
- C. Engineering is installing a temporary modification to the MSL Radiation Monitors. The Switching Order will direct the NPO to remove fuses to de-energize the radiation monitors
- D. Operations is investigating in-leakage into the Suppression Pool. The NPO will open the supply breaker for the Core Spray minimum flow valve, and then manually close the valve. The associated Core Spray Loop has been declared inoperable and a clearance applied to the loop suction and injection valves

From OI-AD-085:

9.10.1 Switching orders are used to control plant evolutions or manipulate plant components when none of the following documents exist to control the evolution:

- Procedures
- Clearance Orders
- Temporary Modification
- Check-off List
- PCWO/RLWO
- PM Activity

9.11.3 The switching order shall receive a 10 CFR 50.59 review (by a qualified individual) in accordance with NDAP-QA-0726, except when:

- a. The switching order is controlling an evolution by coordinating activities controlled by one of the documents identified in 5.1.1, or
- b. The switching order is directing component manipulations of equipment that is out of service and within the boundary of a clearance order.
- c. The switching order is directing component manipulations of equipment that supports the station in performance of maintenance activities such as station hoists or cranes, or any other component manipulations that does not meet the NDAP-QA-0726 requirements for a 10CFR50.59 review

K&A Rating: G 2.2.5 (3.2)

K&A Statement: **Knowledge of the process for making design or operating changes to the facility**

Justification:

- A. **Incorrect but plausible:** This switching order is directing component manipulations of equipment that is out of service and within the boundaries of a clearance order. Per OI-AD-085, this is an exception to receiving a 50.59 review
- B. **Correct:** This switching order does not meet any of the exceptions mentioned in OI-AD-085. Thus, it requires a 50.59 review
- C. **Incorrect but plausible:** This switching order is part of a temporary modification. Per OI-AD-085, this is an exception to receiving a 50.59 review
- D. **Incorrect but plausible:** This switching order is directing component manipulations of equipment that is out of service and within the boundaries of a clearance order. Per OI-AD-085, this is an exception to receiving a 50.59 review

SRO Only Justification: This question is SRO only as requires knowledge of facility licensee procedures required to obtain authority for design and operating changes in the facility

References: OI-AD-085 Rev. 2

Applicant Ref:
None

Learning
Objective:

Question source: SSES bank 131

Question History: Not used on SSES 2008 or 2011 written exams

Cognitive level: Memory/Fundamental knowledge
Comprehension/Analysis: X

10CFR 41.7 / 43.5/45.4

Comments:

SSES 2011 exam question 95 covers similar topic but different question

Question 98

With a liquid effluent discharge in progress RADWASTE EFFLUENT MON HI RADIATION Annunciator (AR-107-E06) alarms on Unit Operating Benchboard 1C601. Abnormally high radiation trends are observed on the Effluent Radiation Recorder RR-06433 on the Liquid Radwaste and Chemical Process Panel OC301.

IAW ON-069-001, Abnormal Radiation Release Liquid, which action should be performed?

- A. Ensure high radiation setpoint countrate calculations are correct and properly entered into RITS-06433 Liquid Radwaste Radiation.
- B. Open the blowdown flow control valve to lower the concentration through additional dilution.
- C. Start an additional River Water Makeup Pump to lower the concentration through dilution.
- D. Ensure the operating release sample tank pump trips.

K&A Rating: 2.3.11 (4.3)

K&A Statement: **Ability to control radiation releases**

Justification:

- A. **Correct:** The LRW Effluent Discharge Isolation valves will automatically isolate upon high radiation signal, and the operators are directed to verify high radiation setpoint countrate calculations and determined if they have been appropriately entered into RITS-06433 Liquid radwaste radiation.
- B. **Incorrect but plausible:** Plausible if the combined cooling tower blowdown low flow caused the LRW effluent discharge isolation, which would require the operators to increase flow greater than 6000 gpm by starting additional pump or via opening flow control valves.
- C. **Incorrect but plausible:** Plausible if the combined cooling tower blowdown low flow caused the LRW effluent discharge isolation, which would require the operators to increase flow greater than 6000 gpm by starting additional pump or via opening flow control valves.
- D. **Incorrect but plausible:** Plausible if the applicant does not understand that the sample pump tank would not be required to be tripped, the operating pump recirculation lines are opened IAW ON-069-001.

SRO Only Justification: This question is SRO only since SRO candidates must assess plant conditions and select the correct procedure or section of a procedure to mitigate or recover the event. This question cannot be answered solely by knowing system knowledge, AOP/EOP entry conditions, or sequence of events/mitigative strategy of a procedure.

References: ON-069-001, Rev. 17

Applicant Ref:
None

Learning
Objective:

Question source: SQ Bank Modified

Question History: Not used on 2011 or 2008 SQ Written

Cognitive level: Memory/Fundamental knowledge X
Comprehension/Analysis:

10CFR 41.11 / 43.4 / 45.10

Comments:

Question 99

A refueling outage is in progress. The following conditions exist:

- Reactor cavity is flooded and the gates removed.
- Reactor coolant temperature is 125°F with RHR shutdown cooling in operation.
- 'B' Reactor Recirculation pump impeller is being replaced.
- Spent fuel moves are occurring in the fuel pool
- Refuel floor personnel are performing the following:
 - LPRM bending activities
 - Control rod blade exchanges

Which activity would require additional access controls to be implemented for the upper drywell elevations?

- A. Control rod blade exchanges
- B. Spent fuel moves in the fuel pool
- C. LPRM bending activities
- D. Recirculation pump impeller replacement

From NDAP-QA-0626:

9.12 Access control requirements for upper elevations of the Drywell, during outages

9.12.1 Access to the upper elevations of the drywell **is permitted** during movement of irradiated components **within** either fuel pool or **within** the cask storage pit provided that none of the following activities is being performed:

- Irradiated fuel movements in the reactor cavity
- Irradiated LPRM handling
- LPRM bending
- Dry tube replacement
- Jet pump inlet mixer movements

K&A Rating: G 2.3.13 (3.8)

K&A Statement: **Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.**

Justification:

- A. **Incorrect but plausible:** Irradiated control rods are a significant radiation source, but access restriction is not required per NDAP-QA-0626
- B. **Incorrect but plausible:** Irradiated fuel is a significant radiation source, but access restriction is not required per NDAP-QA-0626. Access restriction is required, however, for irradiated fuel movements in the reactor cavity
- C. **Correct:** Access to the upper elevations of the drywell is restricted for LPRM bending per NDAP-QA-0626
- D. **Incorrect but plausible:** While the recirculation pump impeller is a significant radiation source, upper drywell restriction is not required per NDAP-QA-0626

SRO Only Justification: This question is SRO only as it requires knowledge of radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

References: NDAP-QA-0626 Rev. 25

Applicant Ref:
None

Learning
Objective:

Question source: Modified SSES bank 190

Question History: Not used on SSES 2008 or 2011 written exams

Cognitive level: Memory/Fundamental knowledge X
Comprehension/Analysis:

10CFR 41.12 / 43.4/45.9/45.10

Comments:

Question 100

The plant is operating at 100% power with the following:

<u>Time (hh:mm)</u>	<u>Event</u>
00:10	Plant conditions justify the declaration of an ALERT
00:17	Plant conditions change and only meet the UNUSUAL EVENT threshold. Emergency declaration has not yet been made.

Which one of the following describes the latest time at which the emergency declaration must be made and the level of emergency that must be declared and reported, in accordance with EP-TP-001, EAL Classification Levels?

	<u>Latest Time For Declaration</u>	<u>Level of Emergency Declaration</u>
E.	00:25	Declare and report the UNUSUAL EVENT
F.	00:32	Declare and report the UNUSUAL EVENT
G.	00:25	Declare and report the ALERT
H.	00:32	Declare and report the ALERT

K&A Rating: 2.4.40 (4.5)

K&A Statement: **Knowledge of SRO responsibilities in emergency plan implementation.**

Justification:

- A. **Incorrect but plausible:** Plausible if the applicant does not recall that per EP-TP-001, if a valid indication of an EAL rapidly exceeds and then decreases below an EAL condition, entry into the emergency plan or escalation to the higher classification is appropriate. Applicant may decide to choose UE based on current plant conditions and latest time calculation would be from the start of ALERT condition.
- B. **Incorrect but plausible:** Plausible if the applicant does not recall that per EP-TP-001, if a valid indication of an EAL rapidly exceeds and then decreases below an EAL condition, entry into the emergency plan or escalation to the higher classification is appropriate. Applicant may decide to choose UE based on current plant conditions and latest time calculation would be from the start of UE condition.
- C. **Correct:** Per EP-TP-001, if a valid indication of an EAL rapidly exceeds and then decreases below an EAL condition, entry into the emergency plan or escalation to the higher classification is appropriate. Valid condition for ALERT was met at 00:10 and latest time calculation would indicate that ALERT must be declared by 00:25.
- D. **Incorrect but plausible:** Plausible if the applicant determines that ALERT is appropriate per EP-TP-001, but latest time calculation would be based upon plant condition change of UE.

SRO Only Justification: This question is SRO only since SRO candidates must assess plant conditions and select the appropriate emergency classification to declare. This question involves diagnostic steps and decision points in the emergency classification procedures.

References: EP-TP-001, Rev. 3

Applicant Ref:
None

Learning
Objective:

Question source: NMP 08/09 Exam

Question History: NMP 08/09 Exam

Cognitive level: Memory/Fundamental knowledge: X

10CFR 41.10 / 43.5 / 45.11

Comments: