PPL SUSQUEHANNA, LLC
STANDARD EXAM SHEET

Course #: ______ Course: LOC24 NRC WRITTEN EXAM

First Name: ______ Last Name: ______ Employee #: ______

Social Security Number

Test Form: A B C D

KEY

Test Date: __/__/__

Test Series: ______

Test Number: ______

Test Taking is an Individual Effort: Any test misconduct is a violation of the Academic Honesty Policy (NTP-QA-14.2) and the PPL Corp. Standards of Conduct and Integrity.

Signature: ______

______ # Correct ______ % Score

* NOTE! following Resolution of Post Exam Comments NRC is Accepting Two Correct Answers to these Questions.

2/21/12

1. B	21. D	41. B	61. C	81. B
2. D	22. A	42. C	62. A	82. B
3. B	23. A	43. B	63. C	83. B
4. B	24. D	44. B	64. D	84. A
5. D	25. D	45. D	65. D	85. B
6. A	26. B	46. C	66. B	*86. C, D
7. C	27. C	47. D	67. B	87. A
8. D	28. C	48. C	68. A	88. D
9. D	29. A	49. C	*69. C, D	89. B
10. D	30. B	50. A	70. B	90. D
11. D	31. C	51. C	71. C	91. C
12. B	32. C	52. D	72. A	92. D
13. B	33. B	53. B	73. C	93. B
14. A	34. B	54. C	74. A	94. C
15. A	35. B	55. B	75. D	95. D
16. B	36. A	56. B	76. C	96. D
17. D	37. D	57. A	77. B	97. B
18. D	38. D	58. C	78. D	98. A
19. C	39. B	59. B	*79. A, D	99. C
20. B	40. D	60. D	80. D	100. C

Question 1

Unit 1 plant conditions are as follows:

- Unit 1 was operating in Mode 1 when 'A' recirculation pump RPT breaker 1A20501 trips, placing the plant in single loop operations
- ON-164-002, Loss of Reactor Recirculation Flow has been entered and operator actions are in progress
- A maintenance worker reports bumping into and tripping the RPT breaker

Current plant conditions:

- Reactor power: 58%
- OPRM Inoperable
- PICSY is unavailable
- 'B' 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 'A' Recirculation Pump to exit the region of the Power/Flow map

B. Place the Reactor Mode Switch in SHUTDOWN

C. Insert control rods in reverse sequence to exit the region of the Power/Flow map

D. Raise the speed of the 'B' Recirculation Pump to 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 due to reactor power at 58% and core flow at 42 Mlbm/hr.

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 ≤80%.

K/A Rating: 295001 Partial or Complete Loss of Forced Core Flow Circulation 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

Units 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 the Montour and Mountain lines.

Assuming all automatic actions occur as designed, and NO operator actions are taken, what is to the status of Aux buses 11A and 11B?

A. Energized by Startup Bus 20 via Startup Transformer 20 (0X104) to Bus 11A and 11B via breakers 1A10104, 1A10204, and tie breaker 0A10502.

B. Energized by Unit Aux Transformer 11 to Bus 11A and 11B via breakers 1A10101 and 1A10201.

C. Energized by Unit Aux Transformer 12 to Bus 11A and 11B via breakers 1A10104, 1A10204, and tie breaker 10502.

D. Both Aux buses are de-energized, tie breaker 0A10502 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 0A10502 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
ON-003-001, Rev. 23 Section 2.11

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

The "D" Diesel Generator is running with its Unit 1 Output Breaker (1A204-04) closed following a valid start signal. 125 VDC Bus 1D644 is then de-energized.

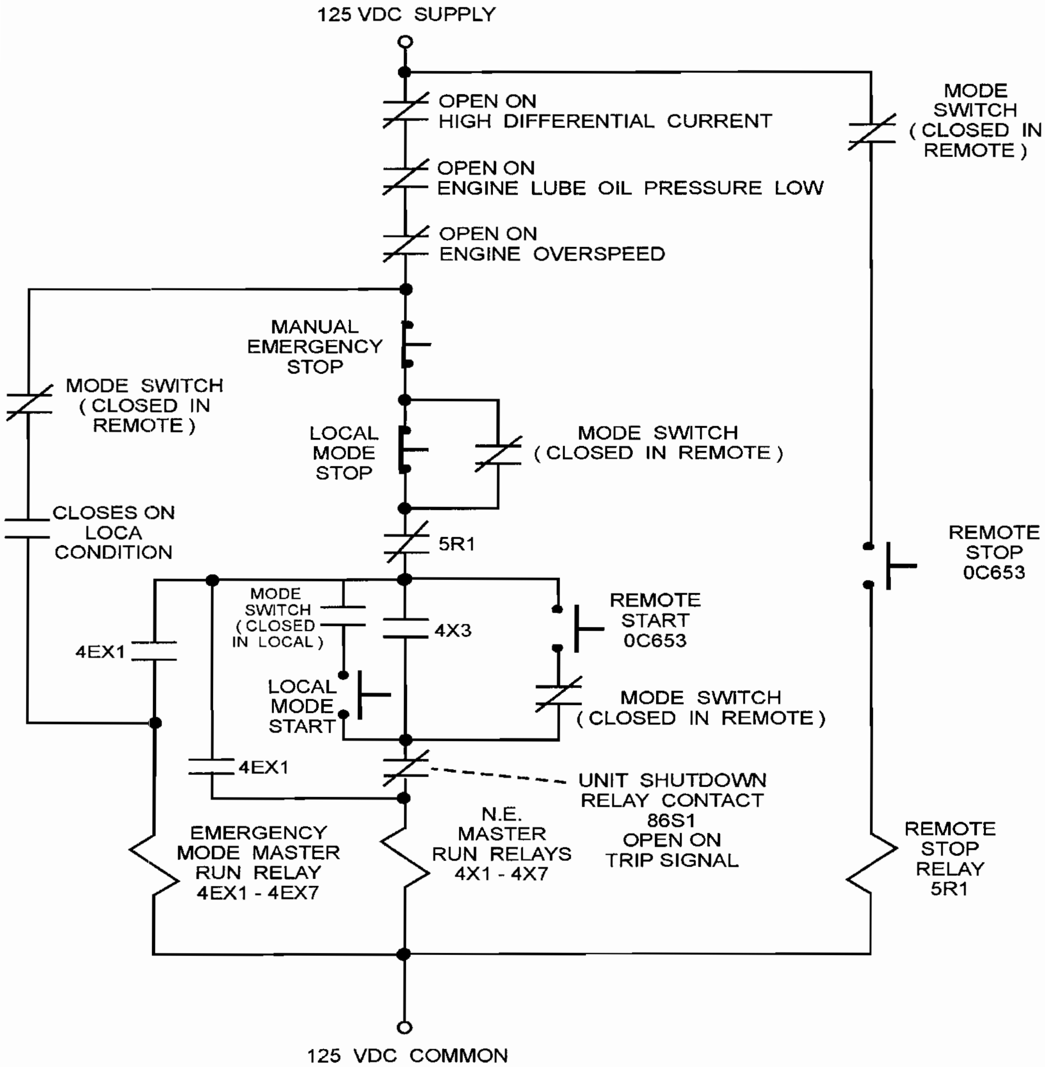
What is (1) the effect on the “D” Diesel Generator, and (2) what action is required by ON-102-640, Loss of 125V DC Bus 1D640?

A. (1) “D” Diesel Generator will trip
(2) verify automatic transfer of “D” Diesel Generator DC power to 2D644

B. (1) “D” Diesel Generator will trip
(2) manually transfer “D” Diesel Generator DC power to 2D644

C. (1) “D” Diesel Generator will continue running, but cannot be restarted if tripped
(2) verify automatic transfer of “D” Diesel Generator DC power to 2D644

D. (1) “D” Diesel Generator will continue running, but cannot be restarted if tripped
(2) manually transfer “D” Diesel Generator DC power to 2D644



125 VDC SUPPLY
OPEN ON
HIGH DIFFERENTIAL CURRENT
OPEN ON
ENGINE LUBE OIL PRESSURE LOW
OPEN ON
ENGINE OVERSPEED
MODE SWITCH (CLOSED IN REMOTE)
MANUAL EMERGENCY STOP
MODE SWITCH (CLOSED IN REMOTE)
LOCAL MODE STOP
MODE SWITCH (CLOSED IN REMOTE)
CLOSES ON LOCA CONDITION
5R1
REMOTE STOP 0C653
MODE SWITCH (CLOSED IN LOCAL)
4X3
REMOTE START 0C653
4EX1
LOCAL MODE START
MODE SWITCH (CLOSED IN REMOTE)
4EX1
UNIT SHUTDOWN RELAY CONTACT 86S1 OPEN ON TRIP SIGNAL
EMERGENCY MODE MASTER RUN RELAY 4EX1 - 4EX7
N.E. MASTER RUN RELAYS 4X1 - 4X7
REMOTE STOP RELAY 5R1
125 VDC COMMON

PNEUMATIC SAFETY SHUTDOWN SYSTEM

(P&ID M-134, Sheet 3)
There are two source paths of air supply to the Pneumatic Safety Shutdown System. One path is provided upstream of the cranking air valves (Control Air) and the other is downstream of the cranking air valves (Reset and Lockout Air).

Control Air

Upstream of the cranking air valves, 250-psig Control Air is directed, via Shuttle Valve PY03447A-D, to Pressure Control Valve PCV03447A-D where it is reduced to 80 psig. This air is then directed to five possible flow paths (Figure 17).

The first path considered is via is the Safety Trip Control Valve PV03431A-D. During diesel operating conditions the valve is positioned to isolate the 80 psig air supply to the downstream components (specifically the actuator of the Fuel Control Cylinder Valve PV03432A-D via shuttle valve PY03483A-D). In series, and downstream of the Safety Trip Control Valve are two normally de-energized solenoid operated Fuel Solenoid Valves SV03423A-D and SV03422A-D. Normally de-energized (125 VDC), these valves are in the open (non-vented) position and will allow air from the Safety Trip Control Valve to shut down the diesel by pressurizing the actuator of the Fuel Control Cylinder Valve PV03432A-D when the Safety Trip Control Valve repositions. During Emergency Mode conditions the Fuel Solenoid Valves energize to prevent the Safety Trip Control Valve from pressurizing the fuel control cylinder and shutting down the diesel

K&A Rating: 295004 AA2.04 Partial or Complete Loss of D.C. Power 3.2/3.3

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

Justification:

A. **Incorrect but plausible:** With a loss of its 125 VDC power source, all control power is lost for the EDG. This results in a loss of power to the master run relays and emergency mode master run relay, causing a loss of pressure to the pneumatic safety shutdown system and an EDG trip. One of the required actions of ON-102-640 is to manually align the alternate Unit 2 control power source to the EDG. Applicant may choose this option if they cannot recall that the control power transfer is manual vice automatic.

B. **Correct:** With a loss of its 125 VDC power source, all control power is lost for the EDG. This results in a loss of power to the master run relays and emergency mode master run relay, causing a loss of pressure to the pneumatic safety shutdown system and an EDG trip. One of the required actions of ON-102-640 is to manually align the alternate Unit 2 control power source to the EDG.

C. **Incorrect but plausible:** While an EDG will not start due to a loss of DC (due to loss of control power and field flash), it will also trip if running. Applicants may not recall that a loss of DC will also cause an EDG trip, and that the control power transfer is manual vice automatic

D. **Incorrect but plausible:** Applicant may choose this option if they do not recall that while an EDG will not start due to a loss of DC (due to loss of control power and field flash), it will also trip if running.

References: ON-102-640, Rev. 10
TM-OP-024-ST, Rev. 10 EDG A-D

Applicant Ref: None

Learning Objective: TM-OP-024-OB Rev. 1: 2072b, 2260g

Question source: Modified SSES Bank 490

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

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

10CFR 41.10/43.5/45.13

Comments:

QUESTION 4

Given the following on Unit 1:

- 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?

Immediately, Reactor PRESSURE will ___(1)___ and Reactor indicated 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, Rev. 10 Page 28 Student Ref: NONE

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, the reactor was manually scrammed, and multiple control rods failed to insert.
- Initial ATWS power was 4%.
- Due to errors in level control, reactor water level reached -57", resulting in SBLC injection for level control.

Current plant conditions are as follows:

- Reactor power is midscale on IRM range 6 and lowering slowly
- Reactor water level is +24" and steady
- Drywell pressure is 1.27 psig and rising slowly

Per 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 the reactor is subcritical due to control rods

D. NOT allowed until Cold Shutdown Boron Weight has been injected

Answer:
A is incorrect. Per EO-100-113 step LQ/P-8, under these conditions if any amount of boron less than Cold Shutdown Boron Weight has been injected, cooldown is not permitted. If cooldown is commenced, recriticality may occur due to the positive reactivity added during cooldown. Applicant may choose this option if they do not recognize that SBLC was injected.

B is incorrect. Per EO-100-113 step LQ/P-8, under these conditions if any amount of boron less than Cold Shutdown Boron Weight has been injected, cooldown is not permitted. If cooldown is commenced, recriticality may occur due to the positive reactivity added during cooldown. Applicant may choose this option if they believe that the reactor has sufficient rods inserted to ensure subcriticality; with more than one rod stuck out, this determination cannot be made without Reactor Engineering assistance.

C is incorrect. Per EO-100-113 step LQ/P-8, under these conditions if any amount of boron less than Cold Shutdown Boron Weight has been injected, cooldown is not permitted. If cooldown is commenced, recriticality may occur due to the positive reactivity added during cooldown. Applicant may choose this option if they do not recognize that SBLC was injected and believe that the reactor has sufficient rods inserted to ensure subcriticality. Since SBLC was injected, it is uncertain whether the reactor is subcritical due to rods alone, or a combination of control rods and SBLC.

D is correct. Per EO-100-113 step LQ/P-8, under these conditions if any amount of boron less than Cold Shutdown Boron Weight has been injected, cooldown is not permitted. If cooldown is commenced, recriticality may occur due to the positive reactivity added during cooldown. If all control rods are not inserted, the reactor cannot be assured to remain subcritical during the cooldown unless either all control rods are inserted or Cold Shutdown Boron Weight has been injected.

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 Applicant Ref: None
EO-100-113 Bases Rev. 8

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

Both Unit 1 & 2 control rooms have been evacuated due to a fire with the following plant conditions:

- The crew was able to complete ALL immediate operator actions per ON-100-009
- Control of Unit 1 has been established at the Remote Shutdown Panel
- RPV pressure is at 600 psig
- RPV level is at +32 inches

Which one of the following describes the response of the U1 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 +18 inches

C. The condensate pumps will need to be started by closing their respective breaker locally, then condensate will inject to the RPV

D. The condensate pumps will NOT inject into the RPV until RPV pressure reaches 575 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:** When RPV pressure drops to ~ 630-640 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 that an ON-100-009 immediate action resets the RPV level setpoint.

C. **Incorrect but plausible:** Plausible if the applicant does not recall that per ON-100-009 immediate actions two condensate pumps will be left in service.

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 at 100 percent power and Unit 2 is shutdown.
The following conditions exist:

- Unit 1 RBCCW System leak is being investigated
- Unit 1 RBCCW Head Tank level is being maintained via normal makeup
- Unit 2 RBCCW System is Shutdown

In accordance with ON-114-001, “LOSS OF RBCCW,” which one of the following actions is required for the given conditions?

A. Investigate leakage into the ESW System

B. Monitor Reactor Recirc Pump A and B motor winding temperatures

C. Investigate leakage into Unit 2 RBCCW via the Low Pressure Air Compressor

D. Monitor Drywell Equipment Drain Tank pumping times for any indication of leakage inside the drywell

Answer:

A is incorrect. While Emergency Service Water can be aligned to provide cooling to RBCCW in the event of a loss of Service Water, under current circumstances Service Water is aligned and leakage into ESW is not plausible.

B is incorrect. ON-114-001 directs monitoring recirc pump motor bearing and seal cavity temperatures, not motor winding temperatures. Recirc Pump motor windings are cooled by Reactor Building Chilled Water.

C is correct. With the Unit 2 RBCCW system out of service, potential exists for leakage into the out of service RBCCW system via the system crosstie valves at the Low Pressure Air Compressor.

D is incorrect. The operators will be monitoring for leakage into the drywell, but leakage will be evidenced by greater influx into the Floor Drain sumps, vice the Equipment Drain Sumps.

K/A Rating: 295018 Partial or Complete Loss of Component Cooling Water 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:	Susquehanna Bank question 593	
Question History:	None	
Cognitive level:	Memory/Fundamental knowledge: Comprehension/Analysis:	X
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) When is the RO required to SCRAM the reactor?
2) What is the basis for this action?

A. 1) 75 psig
2) MSIVs will begin to drift closed, causing a pressure transient and scram.

B. 1) 75 psig
2) Scram Inlet and Outlet Valves will drift open, causing control rods to insert.

C. 1) 65 psig
2) MSIVs will begin to drift closed, causing a pressure transient and scram.

D. 1) 65 psig
2) Scram Inlet and Outlet Valves will drift open, causing control rods to insert.

K&A Rating: 295019 2.4.11 (4.0/4.2)

K&A Statement: **Partial or Total Loss of Inst. Air :** 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: X
Comprehensive/Analysis:

10CFR 41.10/43.5/45.1
3

Comments:

Question 9

Given the following conditions:

- Unit 2 has experienced a loss of Shutdown Cooling while in Mode 3
- Primary and Secondary Containment are established
- The plant has been shut down for 36 hours
- Reactor water level is stable at +48 inches
- Reactor pressure is being maintained 20 to 98 psig by opening Safety Relief Valves (SRV) as needed
- Following opening of one of the preferred SRVs, the PCO is unable to close the valve

Which of the following describes the effect of this failure as the reactor depressurizes?

A. The Acoustic Monitor will become inoperable as discharge downcomer flow lowers

B. The reactor will reach saturation temperature with a significant reduction in the "time-to-boiling" value

C. Adequate core cooling will NOT be maintained for these conditions

D. The SRV tailpipe may begin to reflood with Suppression Pool water

From ON-149-001:

(1) **Maintain** reactor pressure 20 psig to < 98 psig as follows:

(a) **IF** condenser vacuum **ESTABLISHED AND MAINTAINED** by Steam Jet Air Ejector, **Utilize** Bypass Valve Opening Jack,

OR

NOTE:	Reactor pressure must be ≥ 19 psig (Saturation temperature ~ 257°F) to clear column of water from SRV discharge downcomers allowing steam flow.

(b) **Open** one to four non-ADS SRV's in alphabetical order.

In Mode 3 with Primary and Secondary containment established, reactor pressure may be maintained greater than 20 psig (to clear column of water from SRV discharge downcomers if steam flow is routed or diverted there) and less than 98 psig (below reactor high pressure isolation).

K&A Rating: 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**

Justification:

A. **Incorrect but plausible:** Acoustic Monitor operability is not dependent on SRV flow. Applicant may choose this option as acoustic monitor function is affected by the lowering D/P and resultant lowering of flow.

B. **Incorrect but plausible:** Reactor already at saturation (Hot Shutdown), > 200 °F. Applicant may choose this option as time to boil is normally a high visibility item during shutdown operations.

C. **Incorrect but plausible:** Applicant may choose this option if they believe that the reduced reactor pressure will result in insufficient steam flow and insufficient core cooling. With level significantly above TAF, adequate core cooling is assured.

D. **Correct:** Maintaining reactor pressure ≥19 psig is required to ensure the SRV tailpipe does not re-flood due to the static head of water on the SRV spargers. If this is not maintained, steam flow through the SRV will cease, causing a reduction in core cooling due to a loss of steam flow, and reduction in colder feedwater to maintain level.

References: ON-149-001 Rev. 23 Applicant Ref: None

Learning Objective:	AD045: 15308, 15309	
Question source:	SSES bank 651	
Question History:	Not used on SSES 2008 or 2011 written exams	
Cognitive level:	Memory/Fundamental knowledge	
	Comprehension/Analysis:	X
10CFR	41.12 / 43.4/45.9/45.10	

Comments:

QUESTION 10

Unit 1 is in a refueling outage with the following:

- A fuel bundle, which was 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 AR-106-001 (E03) is alarming.

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

A. Ensure both CROEASS fans are running, and evacuate both Reactor and Turbine Buildings.

B. Ensure Zone 3 isolated, and evacuate refuel floor ONLY.

C. Ensure Zone 1 and 3 isolated, and evacuate refuel floor ONLY.

D. Ensure Zone 3 isolated, both SGTS fans are running, and evacuate both Reactor Buildings.

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 only the lead CREOASS fan starts on a Zone 3 isolation and they are not required to evacuate the turbine buildings.

B. **Incorrect but plausible:** Plausible if the applicant does not recall that step 4.4.1 of ON-081-001 directs evacuation of both Reactor Buildings.

C. **Incorrect but plausible:** Plausible if the applicant does not recall that only Zone 3 isolates and step 4.4.1 of ON-081-001 directs evacuation of both Reactor Buildings.

D. **Correct**: Per ON-070-001, one of the automatic action which is needed to be verified is start of the SGTS system, and also step 4.4.1 of ON-081-001 directs evacuation of both Reactor Buildings.

References: ON-081-001, Rev. 10 Applicant Ref: NONE
AR-106-001, Rev. 47

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

The following plant conditions exist after a transient on Unit 1:

- Drywell pressure	9.4 psig and up slow
- Drywell temperature	202 °F and up slow
- Suppression Chamber pressure	8.4 psig and up slow
- Suppression Pool water temperature	82 °F and steady

Which of the statements below 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 discharge 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 High Drywell Pressure (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 EHC Pressure Regulator '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 an 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.

K&A Rating: 295025 EK2.08 (3.7/3.7)

K&A Statement: **Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following:** Reactor/turbine pressure regulating system

Justification:

A. **Incorrect but plausible:** Incorrect, plausible if the candidate does not understand the relationship between the pressure regulator and the signal control of the TCVs.

B. **Correct:** If the output of the "A" pressure regulator fails high, the CVs and BPVs attempt to open until gagged by load limit and MCFL, Rx pressure and power decrease until MSIV isolation occurs on <861 psig in RUN. Rx scrams on MSIV closure.

C. **Incorrect but plausible:** Incorrect, plausible if the candidate assumes that the B channel takes over and causes the pressure to be at a lower value due to B regulator set at a slightly lower value.

D. **Incorrect but plausible:** Incorrect, plausible if the candidate assumes that the B channel takes over.

References: ON-193-001, Rev. 16
TM-OP-093L-FS, EHC Pressure Control and Logic, Rev. 04
TM-OP-093L-OB, LO Systems, EHC Pressure Control and Logic Rev. 01

Applicant Ref: NONE

Learning Objective: TM-OP-093L-OB , R 1641

Question source: New

Question History: None

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

10CFR 41.7/45.8

Comments:

Question 13 (pg 1 of 3)

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

- ATWS
- MSIVs are closed
- Reactor power is 34%, down slow
- Suppression Pool level is 21.5 feet, up slow
- Suppression Pool temperature is 127°F, up fast
- Suppression Pool cooling is not available
- RPV pressure is being maintained 850 to1050 psig using SRVs
- Drywell pressure is 0.67 psig, up slow
- Suppression Chamber pressure is 0.43 psig, up slow

(See attached) Which one of the following limits will be challenged by these conditions?

A. Pressure Suppression Limit

B. Heat Capacity Temperature Limit

C. RHR & CS Vortex Limit

D. Primary Containment Pressure Limit

Question 13, continued (pg 2 of 3)

FIG 2 HCTL

HEAT CAPACITY TEMPERATURE LIMIT

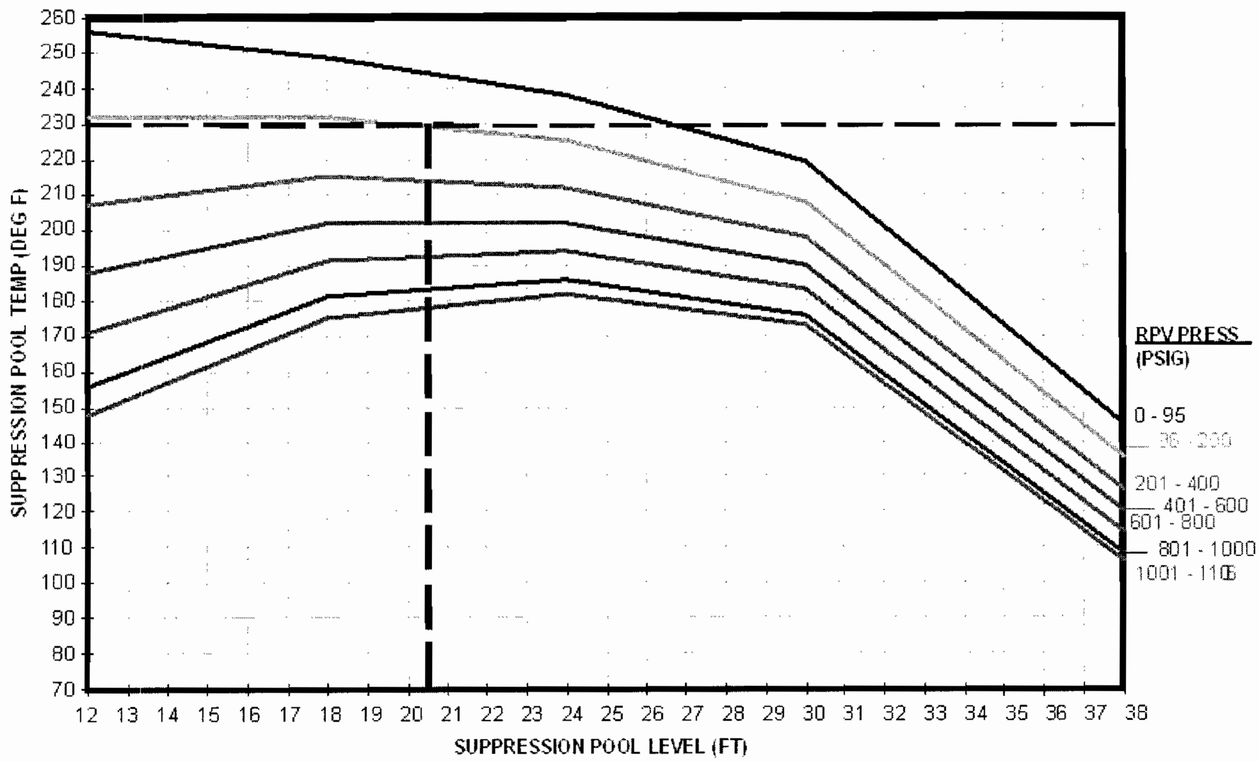
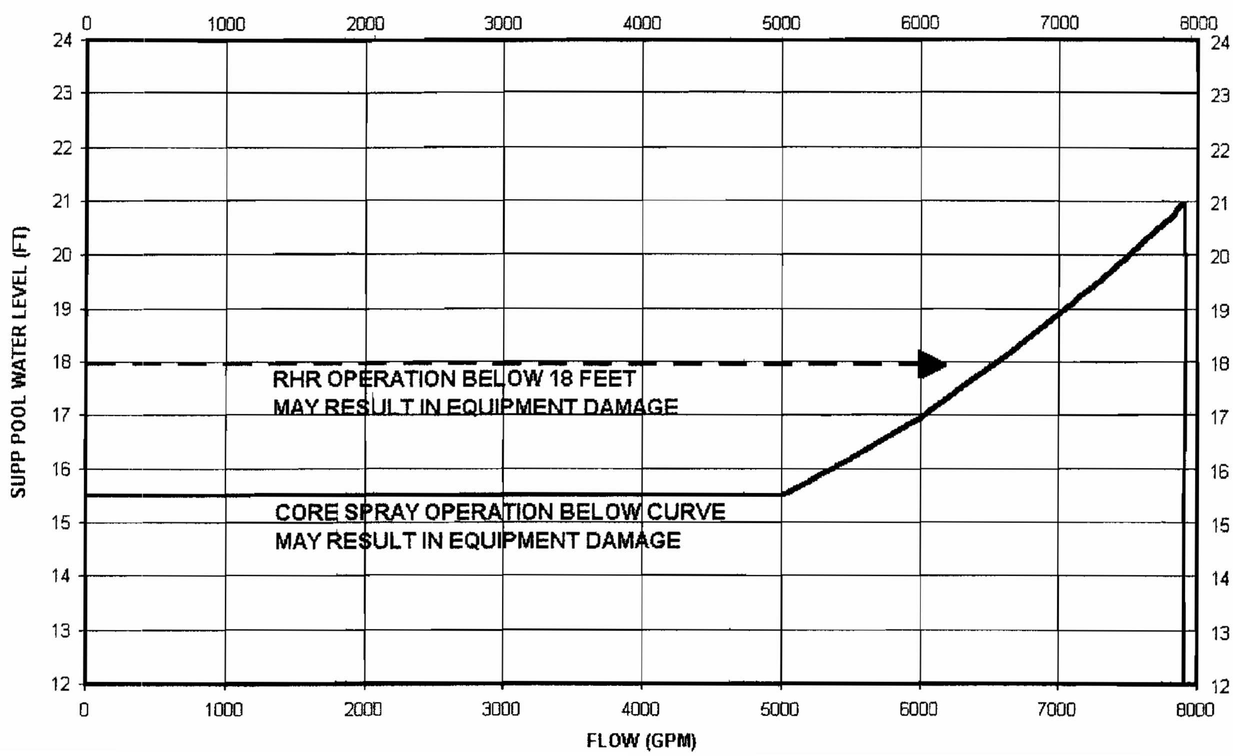
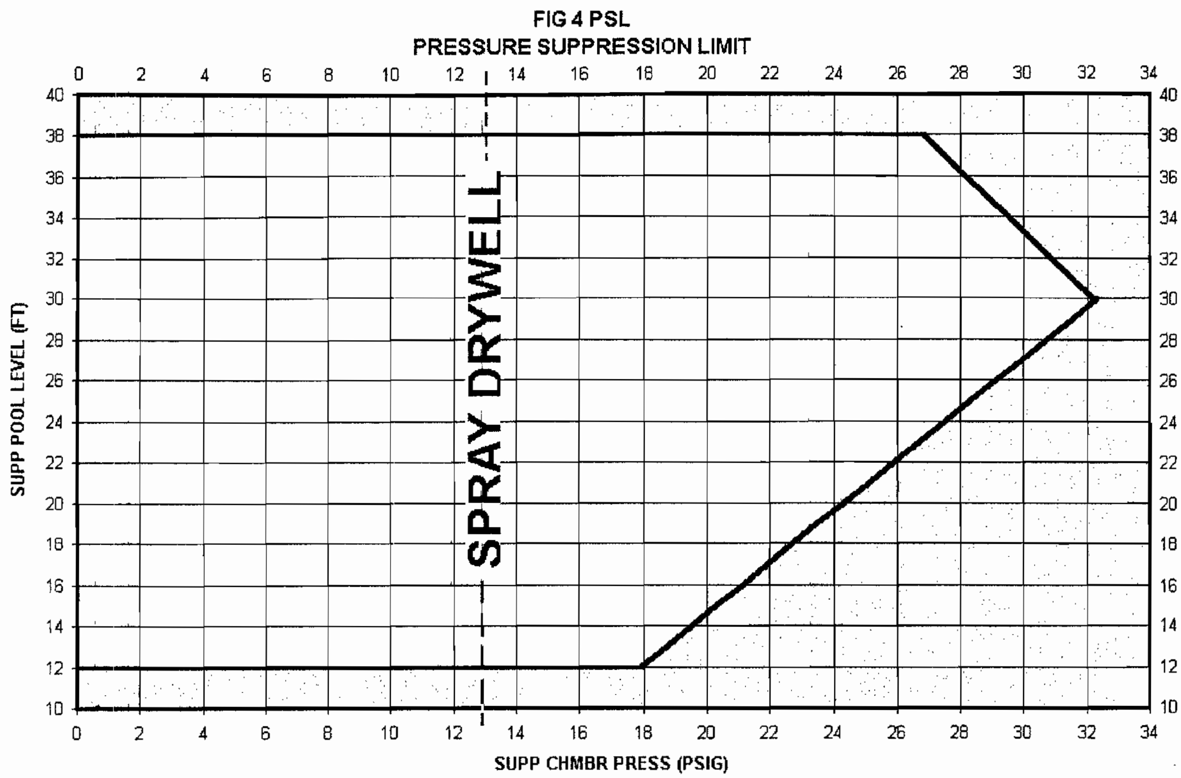


FIG 7 VL

RHR & CS VORTEX LIMIT



Question 13, continued (pg 3 of 3)



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 Suppression Pool High Water Temperature 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: EO-100-103 Graphs

Learning Objective:	TM-OP-059-OB Rev. 1: 10360k
Question source:	PB 12/2009
Question History:	Not used on SSES 2008 or 2011 written exams
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).
- Steam leak inside the Drywell.
- RPV Level is -91 inches; lowering 2 inches/minute.
- RPV Pressure is 720 psig; lowering 10 psig/minute.
- DW Temperature is 250° F; rising at 1° F/minute.
- Drywell Pressure is 18.2 psig; rising.
- Suppression Pool Temperature is 104° F; up slow.
- Suppression Chamber Pressure is 13.1 psig; up slow

Which one of the following EOP actions is required?

A. Initiate Suppression Chamber Sprays followed by Drywell Sprays.

B. Initiate Drywell Sprays, Suppression Chamber Sprays NOT required..

C. Initiate Rapid Depressurization so LPCI can restore RPV level.

D. Depressurize using SRVs and maintain level between +13” and +54” using CRD 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. Also, slow up trend on DW temp does not required direct DW spray initation.

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 that RD criteria have not been met based on basic EOP and system knowledge (DW temp limit and SC parameters).

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 Low Suppression Pool Water Level   EK2.07 (3.5/3.8)
K/A Statement: Knowledge of the interrelations between Low Suppression Pool water level and Downcomer submergence

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. Core Spray flow alone ensures adequate core cooling

B. The combination of Core Spray flow at this reactor water level ensures adequate core cooling

C. Steam Cooling with injection ensures adequate core cooling

D. There is no assurance of 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 candidate thinks that minimum steam cooling is providing adequate core cooling, but incorrect because steam cooling down to -205 is only applicable if there is no injection.

D. **Incorrect but plausible:** Plausible if the applicant does not recall the requirements for adequate core cooling.

References: 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:

- A hydraulic ATWS has occurred
- Reactor Power is currently 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 switches

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 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown 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 Applicant Ref: None
TM-OP-055 Rev. 6

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

Question source: SSES ILO Bank #1055

Question History: None

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

10CFR55 41.7/45.8

Comments:

Question 18

A liquid radwaste discharge is in progress. The following alarm is observed in the Radwaste Control Room:

- Sample Flow Low FAL-06433, AR-RW-004-(B07)

Which one of the following automatic actions, if any, occur?

A. No automatic actions occur

B. Effluent Rad monitor sample pump trips ONLY.

C. Plant Effluent Inbd and Outbd isolation valves HV-06432A1 & A2 close ONLY.

D. Plant Effluent Inbd and Outbd isolation valves HV-06432A1 & A2 close and Effluent Rad monitor sample pump trips.

K&A Rating: 295038 EA1.03 (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 56” and rising. What action is required by ON-013-001 and what is the reason for this action?

	Action	Reason
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 Plant Fire On Site 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.

References: ON-013-001 Rev. 29 Applicant Ref: None

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

Question source: New

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

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

10CFR55 None listed

Comments:

Similar to SSES 2011 written exam question 46

Question 20

Unit 1 is currently at 98% 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. Grid conditions have changed and the generator is now operating at 2 MWe from the 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 terminate the power increase and the controller will default to manual	speed will remained unchanged
D.	will terminate the power increase and the controller will default to manual	will decrease Rx power in ~ 4 MWth increments until sufficient margin has been established

K&A Rating: 700000 AA2.01 (3.5/3.6)

K&A Statement: **Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES:** Operating point on the generator capability curve

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 margin does not become available, then the system would time out at 60 minutes and the controller would default to manual...

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 does not default the other controller to manual.

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
Comprehension/Analysis: X

10CFR 41.7

Comments:

Question 21

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

- Reactor power is 75%
- Total feedwater flow is 12.4 MLB/HR
- A, B, and C Reactor Feedwater Pump flows are 75%

Difficulties in level control cause a slow change in reactor level from +35" to +15".

Current plant conditions are as follows:

- Reactor Power is 38%
- Total feedwater flow is 6.3 MLB/HR
- A Reactor Feedwater Pump flow is 61%
- B Reactor Feedwater Pump flow is 36%
- C Reactor Feedwater Pump flow is 16%

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. Current recirculation pump speed is restricted to 48%. Applicant may choose this if they believe that the 30% limiter is based upon individual feed pump flow
B is incorrect. Current recirculation pump speed is restricted to 48%. Applicant may choose this if they believe they reached the low RPV level setpoint for limiter #1.
C is incorrect. Current recirculation pump speed is restricted to 48%. Applicant may choose this if they believe the reached the low RPV level setpoint for limiter #1 and believed this to be the setpoint for the 48% limiter instead.
D is correct. The 48% limiter is active due to individual reactor feed pump flow <16.4% for the C feed pump flow.

Limiter 1 (30%):
- Reactor low water level (13")
- Total feedwater flow <16.4 percent for >15 seconds (if enabled)
- 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 (if enabled)
- Circ water or condensate pump trip (if enabled)
- Individual reactor feedwater pump flow <16.4 percent (if enabled)
- Manual initiation

K/A: 295009 Low Reactor Water Level 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 continues to rise 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 AA1.02 (3.6/3.6)

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 (1.72 psig). Prior to reaching 1.72 psig, at least two of the sump pumps would have been running due to the 60 gpm leak. Once Drywell pressure reached 1.72 psig and the isolation valves close, the pumps trip after 45 seconds and all four pumps will remain off until the isolation signal is clear and the Drywell Floor Drain System isolation logic is reset.

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 the Drywell floor drain sump IB and OB isolation valves 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). Based on the drywell pressure rate of increase, 1.72 psig was exceeded approximately six minutes ago which would have resulted in the pumps tripping after 45 seconds due to the isolation valves not being full open.

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: Applicant Ref: NONE
TM-OP-76D

Learning
Objective:1928.b

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 maintain Suppression Chamber pressure within the Pressure Suppression Limit. Plant conditions are as follows:

- All control rods are inserted
- Drywell Pressure is 28 psig, up slow
- Drywell temperature is 240°F, up slow
- All high pressure feed sources are unavailable
- Multiple failures prevented injection by LPCI and Core Spray
- RPV is depressurized
- All RPV level indicators trending down slowly
- NRLA 0" NRLB 0" NRLC 1"
- Wide Range A -126" Wide Range B -129"
- Fuel Zone A -129" Fuel Zone B -308"
- Extended Range A -128" Extended Range B -134"

According to ON-145-004, RPV Water Level Anomaly, which RPV level indicators are usable?

A. Wide Range A, Fuel Zone A ONLY

B. Wide Range A/B, Fuel Zone A ONLY

C. Wide Range A, Fuel Zone A, Extended Range A ONLY

D. Wide Range A/B, Fuel Zone A, Extended Range A/B

Answer:
A is correct. MIL for Wide Range for given conditions ~-127.8, MIL for Extended Range ~-110", MIL for Fuel Zone ~-305"
B is incorrect. MIL for Wide Range for given conditions ~-127.8, MIL for Extended Range ~-110", MIL for Fuel Zone ~-305". Wide Range B (-129") is unusable IAW ON-145-004. Applicant may choose this answer if plotting on wrong location of figure 6.
C is incorrect. MIL for Wide Range for given conditions ~-127.8, MIL for Extended Range ~-110", MIL for Fuel Zone ~-305". Extended Range A (-128") is unusable IAW ON-125-004. Applicant may choose this answer if plotting in wrong location of Figure 4 or using 70F curve of Figure 4.
D is incorrect. MIL for Wide Range for given conditions ~-127.8, MIL for Extended Range ~-110", MIL for Fuel Zone ~-305". Wide Range B (-129") is unusable IAW ON-145-004. Both Extended Range A (-128") and Extended Range B (-134") are unusable IAW ON-145-004. Applicant may choose this answer if plotting in wrong location of Figures 4/6 or using 70F curve on either figure.

K/A: 295012 High Drywell Temperature AK1.01 3.3/3.5
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: ON-145-004 Attachment C **<u>WITHOUT</u>** table on pg. 21

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

Unit 1 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. Inadvertent isolation of the Reactor Water Cleanup System

D. An Extraction Steam Bleeder Trip Valve closes on high feedwater heater level

K&A Rating: 295014AA2.03 (4.0/4.3)

K&A Statement: **Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION:** Cause of reactivity addition.

Justification:

A. **Incorrect but plausible:** Incorrect but plausible the circulating water pump trip will cause an increase in condensate temperature and therefore a higher feedwater temperature and a lower power level.

B. **Incorrect but plausible:** Incorrect but plausible if the applicant does not understand that the CVs and BPVs will attempt to open until gagged by Load Limit and MCFL pressure and power decrease until MSIV Isolation occurs on <861 psig in RUN. If the pressure regulator fails low then the power would increase due to pressure increase from result of 'B' pressure regulator taking over and closing CVs slightly.

C. **Incorrect but plausible:** the reactor power level will be lower due to not having to heat the cleanup flow return to the reactor.

D. **Correct:** Correct loss of extraction steam due to closure of the bleeder trip valves on high FW heater level will have a major affect on the efficiency of the feedwater heaters. This will cause colder feedwater to enter the reactor causing reactor power to increase. Note the high level also closes the extraction steam isolation valve

References: ON-156-001 Applicant Ref: NONE
TM-OP-047-ST, FW Heater and Extraction Steam, Rev. 06

Learning Objective: 14586, Ops procedure lesson plan

Question source: NMP 09.

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 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 2238, 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
- Unit 2 is currently in a refueling outage with CRD system out of service

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?

A. (1) Perform ON-100-101, SCRAM, SCRAM IMMINENT ONLY
(2) 2302

B. (1) Perform ON-100-101, SCRAM, SCRAM IMMINENT ONLY
(2) 2304

C. (1) Place the mode switch in SHUTDOWN
(2) 2302

D. (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 Loss of CRD Pumps 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 Applicant Ref: None
ON-155-007 Rev. 21

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

Question source: Modified PB 12/2008

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 Leak Detection Hi Temp/Hi Diff Temp AR-108-001 (E05) goes into alarm.
- RCIC Equipment Room Area Temperature reading is 120°F, up slow.
- A steam leak in the RCIC room has been confirmed.
- EO-100-104, "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.

RCIC Room Temperature is now 150 °F, up slow. At this time, what is the plant's response and required operator action?

	Plant Response	Required Operator action
A.	RCIC will automatically isolate	Re-establish Reactor Building HVAC
B.	RCIC will NOT automatically isolate	Start SGTS Fan 0V109B
C.	RCIC will automatically isolate	Start SGTS Fan 0V109B
D.	RCIC will NOT automatically isolate	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 need to 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

Unit 2 was operating at 100% power when all condensate pumps tripped, causing a reactor scram. The following conditions exist:

- The watertight door between HPCI and RCIC was left open with a hose running through it for maintenance support
- AR-214-001 (H03) HPCI Pump Room Flooded alarm is in
- AR-208-001 (H03) RCIC Pump Room Flooded alarm is in
- NPO dispatched to the scene reports a steam/water leak upstream of 255F005, HPCI discharge check valve and reports observation from 670' elevation that HPCI/RCIC room levels are approximately 36" and rising slowly
- Attempts to close HPCI Injection and Pump Discharge valves HV255F006 and HV255F007 have been unsuccessful

Using the included table, which of the following actions is required by EO-200-104, Secondary Containment Control?

TABLE 10
REACTOR BUILDING WATER LEVEL

RB AREA (645 FT EL)	MAX NORMAL WATER LVL	MAX SAFE WATER LVL (IN)
HPCI EQUIPMENT AREA	**HI ALARM**	**26**
RCIC EQUIPMENT AREA	**HI ALARM**	**23**
RHR PUMP ROOM A	**HI ALARM**	**84**
RHR PUMP ROOM B	**HI ALARM**	**86**
CS PUMP ROOM A	**HI ALARM**	**24**
CS PUMP ROOM B	**HI ALARM**	**23**

A. Continue plant shutdown in accordance with GO-200-004, Plant Shutdown to Minimum Power

B. Initiate cooldown ≤100 °F/hr in accordance with EO-200-102, RPV Control

C. Perform Rapid Depressurization in accordance with EO-200-112, Rapid Depressurization

D. Open all bypass valves irrespective of cooldown rate in accordance with EO-200-102, RPV Control

Answer:

A is incorrect. A primary system is discharging into the secondary containment area and it is the cause of the Max Safe Condition.

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 (HPCI Room) 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 Secondary Containment High Sump/Area Water Level EK3.01 2.6/2.8
K/A Statement: Knowledge of the reasons for emergency depressurization as it applies to Secondary Containment High Sump/Area Water Level

References:	EO-200-104 Bases Rev. 6	Applicant Ref: None

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-200-112, "Rapid Depressurization" is in progress with all 6 ADS Safety Relief Valves open. Current conditions are as follows:

- RPV level	-170 inches, up slow
- RPV pressure	390 psig, down slow
- Drywell pressure	10 psig, stable
- 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:** if the applicant does not recall RHR will not begin to inject until approx. 280 psig.

B. **Incorrect but plausible:** if the applicant does not recall RHR will not begin to inject until approx. 280 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 until approx 300 psig.

D. **Incorrect but plausible**: if the applicant does not recall RHR will not begin to inject until approx. 280 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: Peach Bottom bank.

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:

- 1B RHR pump in shutdown cooling
- All other low pressure ECCS in a standby alignment
- A bus lockout of ESS Bus 1B occurs
- No actions have been taken in ON-104-202, "Loss of 4kV ESS Bus 1B"

Which RHR pump(s) and associated flowpath(s), if any, are capable of being lined up to provide shutdown cooling?

A. None

B. 1D RHR Pump ONLY

C. 1A and 1C RHR Pump ONLY

D. 1A, 1C, and 1D RHR Pump ONLY

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 Shutdown Cooling System (RHR Shutdown Cooling Mode) 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 and flow continually lower to 0 rpm/gpm, respectively.

C. HPCI speed and flow remain the same.

D. HPCI trips due to a loss of speed reference signal.

K&A Rating: 206000K4.09 (3.8/3.9)

K&A Statement: **Knowledge of HIGH PRESSURE COOLANT INJECTION SYSTEM design feature(s) and/or interlocks which provide for the following:** Automatic flow control: BWR-2,3,4

Justification:

A. **Incorrect but plausible:** Incorrect but plausible if the applicant does not understand the relationship between low value gate and TCV.

B. **Correct:** The ramp generator and the flow error signals feed into a LVG. The lower of the two signals is passed to the turbine control valve. If the ramp generator fails low, this low signal will pass to the TCV and it will close causing speed and flow to drop to 0.

C. **Incorrect but plausible:** Plausible if the applicant does not understand the relationship between low value gate and TCV.

D. **Incorrect but plausible:** Plausible if the applicant does not understand the relationship between low value gate and TCV.

References: SO-152-002, Rev. 47 Applicant Ref: NONE
TM-OP-052-ST, HPCI, Rev. 05

Learning Objective: 2037, TM-OP-052-OB, LO Systems, High Pressure Coolant Injection

Question source: New.

Question History: None

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

10CFR 41.7

Comments:

Question 31

Given the following conditions:

- An ATWS is in progress on Unit 1
- Reactor Power is 21%
- RPV Water Level is being maintained between -60" and -110"
- Power is lost to 480V panel 1B216
- AR-107-001 (A03) SBLC SQUIB VALVES LOSS OF CKT CONTINUITY alarm is in
- Both squib valve status lights are extinguished
- 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 1A is available
(2) Initiate SBLC per OP-153-001, "Standby Liquid Control System"

B. (1) ONLY SBLC Pump 1B 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:

2. <u>PROBABLE CAUSE</u>:

- 2.1 System initiation.
- 2.2 Blown control power fuses F1A, F2, F3, F4, F5, F6, F23, or F24 for squib valves at Panel 1C617 in Upper Relay Room.
- 2.3 Decomposition of igniter elements in either explosive valve.
- 2.4 Loss of 'A' Squib Firing Power 1Y23621.
- 2.5 Loss of 'B' Squib Firing Power 1Y21607.
- 2.6 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
- 1D622-012, ESS 125 VDC Distribution Panel 1D624 Supply breaker tripped.

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 Spray Loop	**'B' Core Spray Loop**
A.	Both pumps ON; HV152F005A is OPEN	Both pumps ON; HV152F005B is OPEN
B.	Both pumps ON; HV152F005A is CLOSED	Both pumps OFF; HV152F005B is CLOSED
C.	Both pumps ON; HV152F005A is OPEN	Both pumps OFF; HV152F005B is CLOSED
D.	Both pumps OFF; HV152F005A is CLOSED	Both pumps OFF; HV152F005B is CLOSED

K&A Rating: 209001K6.04 (2.8/2.9)

K&A Statement: **Knowledge of the effect that a loss or malfunction of the following will have on the LOW PRESSURE CORE SPRAY SYSTEM :** D.C. power

Justification:

A. **Incorrect but plausible:** Incorrect but plausible if the applicant thinks that logic channel provides pump start signal to BOTH loops.

B. **Incorrect but plausible:** Incorrect but plausible if the applicant does not recall the injection valve interlock. The injection valve auto opens at Hi DW Pressure (1.72 psig) and Low RPV Pressure (420 psig).

C. **Correct:** Correct, Both pumps on loop A will start based on LOCA signal and injection valve will be open due to DW pressure and RPV pressure logic satisfaction. However, on Loop B the pumps will not start due to loss of 125 VDC to DIV II CS logic, and the associated injection valve will be closed.

D. **Incorrect but plausible:** Incorrect but plausible if the applicant thinks that logic channel provides pump start signal to BOTH loops.

References: TM-OP-051, Rev. 3
AR-157-001_5
TM-OP-051-OB, LO Systems, Core Spray Rev01, 2093 & 2080

Applicant Ref: NONE

Learning Objective: 2093, TM-OP-051-OB, LO Systems, Core Spray

Question source: New.

Question History: None

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

10CFR 41.7/45.7

Comments:

Question 33

Unit 1 has just experienced a loss of level control and reactor scram. Plant conditions are as follows:

- AR-109-001 (A04) ECCS Loop A RX LO LEVEL alarm is in
- AR-113-001 (A04) ECCS Loop B RX LO LEVEL alarm is NOT in
- HPCI is out of service for Aux Oil Pump maintenance
- RCIC is injecting at full flow in Automatic
- RPV pressure is 940 psig, steady
- RPV water level is -126", down slow
- RHR Pump A is running
- Core Spray Pump A is running
- Division 1 Drywell Area Unit Coolers have tripped
- EDGs A and C are running unloaded

Based upon the above conditions, what actions are required by the operator?

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

B. Start RHR Pumps B, C, D, Core Spray Pump C, and D EDG

C. Start RHR Pump C and Core Spray Pump C, ONLY

D. Inhibit Div I ADS

Answer:
A is incorrect. Applicant may choose this if they believe that only Div I equipment should be running due to only the A loop ECCS alarm in. Although due to crossties in logic, all RHR pumps and the D EDG should also be running

B 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
- Div 1 Drywell Area Unit Coolers trip

Additionally, due to logic crossties, RHR pumps B and D start, and also D EDG starts

C is incorrect. C RHR and C Core Spray pumps should be started. Due to logic crossties, D EDG should also 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.

D 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. Guidance is also provided in AR-110-001 (A01) to inhibit ADS via EOP or if they believe the alarm to be erroneous.

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:
Comprehension/Analysis: X

10CFR55 41.10/43.5/45.3

Question 34

Unit 1 was manually scrammed from 100% power due to a loss of vacuum.

Current plant conditions are as follows:

Reactor power is still at 36%
RPV level is +25 inches and steady
RPV pressure is 950 psig and is being controlled via SRVs.

The Unit Supervisor directed initiation of Standby Liquid Control and the RO placed HS-14804 SBLC MANUAL INITIATION keylock switch to A START.

Under which one of the following conditions is the SLC system 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 Discharge Pressure is 1,150 psig 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 and B White indicating lights are lit.

D. SBLC Pump Discharge Pressure is 1,520 psig and both SBLC SQUIB READY A and 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. **Correct:** SBLC pump A discharge pressure of 1150 and SBLC SQUIB READY A White indicating light is extinguished indicates a successful injection SBLC system. However, the system did not respond correctly, indicating light for SBLC SQUIB READY B white light did not extinguished.

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. **Incorrect but plausible:** Pump discharge pressure (1520) is high indicating that there is a blockage somewhere in the system and the relief valve is open. NO flow is being directed to the vessel. The discharge relief setpoint is 1500 psig.

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

Applicant Ref: NONE

Learning Objective: 1210.B Predict the Standby Liquid Control System response to manipulation of the following controls.

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
- 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 transformer lockout
- Breaker 1A20109 failed to automatically close

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

A. NO HALF or FULL Scram

B. HALF Scram on RPS ‘A’, ONLY

C. HALF Scram on RPS ‘B’, ONLY

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 Reactor Protection System 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 Applicant Ref: None
TM-OP-004-ST Rev. 2

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

Question source: Modified PB 1/2011 exam

Question History: Not used on 2008 or 2011
Susquehanna written exam

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

10CFR55 41.7

Comments:

Question 36

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

- APRM CHANNEL '1' is reading five percent.
- APRM CHANNEL '2' is reading five percent.
- APRM CHANNEL '3' is reading six percent.
- APRM CHANNEL '4' is reading four percent.
- IRM CHANNEL 'A' is reading 105 on Range 8.
- IRM CHANNEL 'B' is reading 119 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 ONLY.

B. Rod block and full scram.

C. Rod block and RPS B half scram.

D. No automatic actions should occur.

K&A Rating: 215003 K1.01 (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. Applicant may determine 'B' RPS half scram based on IRM channel 'B' reading of 119.

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 Shutdown Margin testing 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 being withdrawn and is currently at position 12
- All other control rods are at 00
- AR-104-001 (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 cannot be inserted or withdrawn

B. No scram signals are generated and control rod 14-19 stops moving and CANNOT be withdrawn

C. 'B' RPS half scram signal is generated and control rod 14-19 stops moving

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 (setpoint is set at 1×10^5) will cause a full reactor scram. Control rod 14-19 is rapidly inserted into the core under scram conditions.

K/A: 215004 Source Range Monitor System A3.03 3.6/3.5
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 Applicant Ref: None
TM-OP-078A-ST Rev. 4

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:

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-103-001 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 in the current condition.

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 an 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 that to bypass #3 would require unbypassing #1. Only one APRM channel is allowed to be bypassed at a time...

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 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 Average Power Range Monitor/Local Power Range Monitor System A3.08 3.7/3.6
K/A Statement: Ability to monitor automatic operations of the Average Power Range Monitor/Local Power Range Monitor System including control rod block status

References: TM-OP-052-ST Rev. 0 Applicant Ref: None
EO-000-103 Bases Rev. 8
AR-114-E01 Rev. 23

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. 22 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 reactor power is 56% power and steady. The following conditions now exist:

- AR-106-001 (A12), 125 VDC Panel 1L610 System Trouble, is in alarm
- NPO confirms 125 VDC Distribution Panel 1D614 de-energized and unable to be restored
- A reactor scram has just occurred due to EHC pressure swings

Based upon the above conditions: (1) What is the effect on RCIC? and (2) What actions must be taken if RCIC is needed for level control?

A. (1) RCIC will initiate automatically at -30 inches and align for injection
(2) Verify automatic initiation by performing section 2.2, "Automatic Initiation" of OP-150-001, "RCIC System"

B. (1) RCIC will not initiate automatically or by use of manual pushbutton
(2) Perform section 2.5, "Manual Startup Using Turbine Trip and Throttling Valve" of OP-150-001, "RCIC System"

C. (1) RCIC is unavailable with no power to 125 VDC distribution panel 1D614
(2) Verify automatic initiation of HPCI by performing section 2.2, "Automatic Initiation" of OP-152-001, "HPCI System"

D. (1) RCIC will not initiate automatically or by use of manual pushbutton
(2) Perform section 2.4, "Manual Component By Component Startup " of OP-150-001, "RCIC System"

Answer:
A is incorrect. Loss of 1D614 causes a loss of inverter power, and loss of automatic/manual pushbutton initiation. Applicant may choose this distractor if they do not recall that a loss of 1D614 causes a loss of automatic/manual pushbutton initiation.

B is correct. Loss of 1D614 causes a loss of inverter power, and loss of automatic/manual pushbutton initiation. A failure of the inverter will result in the loss of the Turbine Control System and the Turbine Governor (Control) Valve (FV-15013) failing to the "FULL OPEN" position. System operation is still possible under these conditions by manually controlling the position of the Turbine Trip and Throttle (Stop) Valve (HV-15012).

C is incorrect. Loss of 1D614 causes a loss of inverter power, and loss of automatic/manual pushbutton initiation. Operation of RCIC is still possible by using section 2.5 to startup the RCIC turbine using the turbine trip and throttle valve. Applicant may choose this distractor if they believe that the loss of 1D614 disables all operation of RCIC, as it in fact disables all modes of startup except one.

D is incorrect. Loss of 1D614 causes a loss of inverter power, and loss of automatic/manual pushbutton initiation. While it does cause a loss of automatic/manual pushbutton initiation, operation of RCIC is still possible by using section 2.5 to startup the RCIC turbine using the turbine trip and throttle valve. Using section 2.4 for manual component by component startup will result in a turbine overspeed condition as the loss of inverter leads to the turbine governor valve traveling to the full open position. Applicant may choose this distractor if they do not recall that the loss of governor will lead to an overspeed condtion, as it is very similar to the section for startup using the turbine trip and throttle valve

K/A: 217000 Reactor Core Isolation Cooling System 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 ON-102-610 Rev 12 Att	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 post 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?

A. Depress both ADS Logic Timer Reset pushbuttons to reset both timers.

B. Inhibit ADS in accordance with OP-183-001.

C. No actions necessary due to automatic ADS logic reset.

D. Ensure ADS automatically initiates in 12 seconds.

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
TM-OP-83E-ST, Automatic Depressurization System, Rev. 0

Applicant Ref: NONE

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 the ‘B’ Channel ambient temperature instrument (Riley Module TSH-G33-1N600B) is upscale, but all other instruments are reading normally.

30 seconds after the alarm, when you observe RWCU system status, it is operating 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 the isolation is delayed for 45 seconds

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, and should have occurred after one second. Alarms are provided, but they are provided at a lower setpoint by separate equipment, and as a warning that an isolation may be forthcoming. Applicant may confuse this isolation with other isolations for the RWCU that have time delay isolations (high delta flow, high system flow)
From AR-101-001 (A03):

3. AUTOMATIC ACTION:

IF condition exists for one second:

- ☐ 3.1 HV-144F004 RWCU INLET OB ISO **CLOSES**.
- ☐ 3.2 **WHEN** HV-144F004 leaves open position, 1P221A & 1P221B, RWCU RECIRC PUMP TRIP.
- ☐ 3.3 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-106-001 (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 initiation capability from 1C601 is lost, however, the SRV handswitches will still function.

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 SRV and non-ADS SRV handswitches on 1C601 will not function on a loss of 1D614.

D. **Incorrect but plausible:** Plausible if the applicant does not recall that it is the SRV function, not the ADS function that is inoperable on the loss of 1D614.

References: TM-OP-083-ST, Main Steam System, Rev. 10. Applicant Ref: NONE
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 immediately on high reactor water level

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 remain at approximately the same values

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.

<u>RFPT Level 8 trip (+ 54 inches)</u>
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: None

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

Question source: New

Question History: None

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

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, III) and the Standby Gas Treatment (SGT) System?

	Zone I	**Zone II**	**Zone III**	**SGT Fan "A"**	**SGT Fan "B"**
A.	Does **NOT** isolate	Does **NOT** isolate	Isolates	Starts	Starts
B.	Isolates	Isolates	Does **NOT** isolate	Starts	Remains Off
C.	Does NOT Isolate	Isolates	Isolates	Starts	Starts
D.	Does NOT Isolate	Isolates	Isolates	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 II and III 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 Unit 1 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 SUB-10 lockout occurs
- At T=15 seconds, a DBA LOCA occurs

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

A. Pumps A, B, and D will start at T=25.5 seconds ONLY

B. Pumps A, B, and D will start at T=30 seconds ONLY

C. Pumps A, B, C, and D will start at T=25.5 seconds

D. 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 (1A201 and 1A203 transfer to their alternate power supply)

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.

- 'A' Diesel is synchronized to the 1A ESS bus per the SO-024-001A and is loaded to 4,000 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 'A' EDG and its output breaker, 1A201-04, and
(2) manual actions required, if any?

A. (1) Running in Emergency Mode with 1A201-04 closed
   (2) NO manual actions required

B. (1) Tripped and 1A201-04 is open
   (2) Start it in Isochronous and then close 1A201-04

C. (1) Running in Non-Emergency Mode with 1A201-04 closed
   (2) Manually separate it from bus 1A201

D. (1) Tripped and 1A201-04 is open
   (2) After it restarts in Emergency Mode manually close 1A201-04

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 determines that the EDG will trip and restart automatically in emergency mode with its output breaker closing automatically.

B. **Incorrect but plausible:** Plausible if the applicant believes that the EDG will trip on overexcitation or underfrequency and the EDG will be required to be restarted in emergency mode and the output breaker will be required to be manual action to 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:
At T=0

- 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

At T+10 hours, which of the following describes an instrument that is ultimately affected by these conditions?

A. Source Range Channel 'A'

B. RBCCW Radiation Monitor

C. Intermediate 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 1 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 open.

B. Both instrument air compressors are running fully loaded, the SA cross-tie pressure control valve is closed.

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 SA cross-tie pressure control valve is closed.

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. Partially correct, that both instrument air compressors will be running, however, service air cross tie should be open due to the system pressure. SA crosstie valve will start to open at 95 psig and be fully opened at 90 psig.

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. Both IA compressors should be running, and service air cross tie should be open due to the system pressure. SA crosstie valve will start to open at 95 psig and be fully opened at 90 psig.

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 <u>disconnected from the grid</u>	'A' EDG operating <u>in parallel with the grid</u>
A.	'A' EDG output voltage remains the same KVARs lower	Bus voltage lowers 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: Nine Mile Point 11/2010

Question History: None

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

10CFR55 41.7/45.5 to 45.8

Comments:
New

Question 52

Which feature allows a control rod drive mechanism to be removed from the reactor for repair with coolant in the RPV?

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 (3.0/3.0)

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, assuming NO operator action?

A. Both 1A and 1B RBCCW Pumps are running

B. 1B RBCCW Pump is running and 1A RBCCW Pump is NOT running

C. Both 1A and 1B RBCCW Pumps are NOT running

D. 1A RBCCW Pump is running and 1B RBCCW Pump is NOT running

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: Not used on SSES 2008 or 2011 written exam

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

10CFR55 41.7/45.7

Comments:

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. In addition:

- Reactor Pressure is now 320 psig, down slow
- Reactor Level reached -40" and is now 37", steady
- Drywell Pressure is 12 psig, up slow

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 Motor breakers OPEN.

C. Tripped with RPT breakers OPEN and Recirculation MG Drive Motor breakers OPEN.

D. Tripped with their Recirculation Pump Discharge and Discharge Bypass Valves closed.

K&A Rating: 202001 K3.05 (3.7/3.9)

K&A Statement: **Knowledge of the effect that a loss or malfunction of the RECIRCULATION SYSTEM will have on following:**
Recirculation system MG sets

Justification:

A. **Incorrect but plausible:** Plausible if the applicant does not see that level has decreased below level 2, which would initiate an ATWS RPT breaker trip. If the level did not go below -38 inches than the pumps would have run back to min (30 percent). When ATWS RPT breakers trip, it also opens the drive motor breakers, so the pumps would not be running.

B. **Incorrect but plausible:** Plausible if the applicant does not recall that the ATWS RPT breaker trip would occur due to reactor low level 2 (-38 inches) and ATWS RPT breaker trip would also open the drive motor breakers. Drive Motor breakers do have an automatic trip features without tripping the RPT breakers.

C. **Correct:** ATWS RPT breaker trip due to reactor low level 2 (-38 inches) and ATWS RPT breaker trip would also trip the Recic MG Drive motor breaker.

D. **Incorrect but plausible:** Plausible if the applicant does not recall the low pressure permissive to close the valves (236 psig).

References: TM-OP-064C-ST,Reactor Recirculation System, Rev. 10   Applicant Ref: NONE

Learning Objective: 2558, LO Systems, Reactor Recirc

Question source: SQ Bank modified.

Question History: Not used on 2011 or 2008 SQ Written

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

10CFR 41.7 / 45.4

Comments:

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: 201006 Rod Worth Minimizer System A2.07 2.5/2.8
K/A Statement: Ability to (a) predict the impacts of RWM hardware/software failure 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: Not used on SSES 2008 or 2011 written exams

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

10CFR55 41.5/45.6

Comments:

Question 56

While performing a rod pattern adjustment, the PCOM continuously inserted Rod 22-35 from Position 48 to 32; then releases the Insert pushbutton. A few seconds later, the ROD DRIFT annunciator is received.

To evaluate the situation, the PCOM presses the following "Display Select" pushbuttons for the full core display, and notes the corresponding indications on the full core display for Rod 22-35:

Display Select <u>Pushbutton</u>	Corresponding Indication <u>Lamp Status</u>
Display Rods Drifting	Red Lamp Illuminated
Display Scram Valves Open	Red Lamp Not Illuminated

Which of the following describe the above indications for Rod 22-35 (No other alarms are received):

A. the Scram Inlet Valve is open, the control rod is partially inserted.

B. the Scram Outlet Valve is open, the control rod is fully inserted.

C. the Scram Inlet and Outlet Valves are open, the control rod is fully inserted.

D. the Insert Rod pushbutton fails to release, the control rod is partially inserted.

K&A Rating: 201002 K1.01 (3.2/3.2)

K&A Statement: **Knowledge of the physical connections and/or cause effect relationships between REACTOR MANUAL CONTROL SYSTEM and the following:** Control rod drive hydraulic system.

Justification:

A. **Incorrect but plausible:** Plausible if the applicant does not recall scram inlet valve opening may also drive rod in, however, it would also result in accumulator fault/trouble alarm.

B. **Correct:** Provides vent path off top of CRDM operating piston, and rod would drift full in with reactor pressure.

C. **Incorrect but plausible:** Plausible if the applicant does not recall that both scram inlet and outlet valves opening provides normal scram water flowpath, however, with that configuration you would also get accumulator fault and scram valves light.

D. **Incorrect but plausible:** Plausible if the applicant does not recall that if the rod pushbutton fails to release, you would not get a drift alarm as the bus remains energized, preventing the drift alarm.

References: TM-OP-056A Applicant Ref: NONE

Learning Objective: 2465, TM-OP-056A-OB, LO Systems, Reactor Manual Control System

Question source: Bank modified.

Question History: Not used on 2011 or 2008 SQ Written

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

10CFR 41.2 - 41.9 / 45.7 / 45.8

Comments:

Question 57

Unit 1 is starting up after a 6 month outage, and:

- Reactor is critical in the source range
- RPV temperature is 126 °F up slow
- RPV water level is being controlled by RWCU in the automatic letdown mode
- Feedwater Startup Bypass Valve HV-10640 is currently closed
- Condensate is in long path recirculation mode

Which of the following describes the response of reactor water level if the running CRD pump trips?

A. Remains stable at approximately 35”

B. Lowers to approximately 33” and stabilizes

C. Lowers to approximately 30” and stabilizes

D. Lowers to approximately -38” and stabilizes

Answer:

A is correct. With the loss of water addition from the CRD pump, reactor water level will lower continuously from 35" due to letdown flow. Due to the lack of decay heat and very slowly lowering level, the letdown valve controller will have sufficient time to close the level control valve before significant inventory loss occurs. This is also confirmed with simulator modeling; water level dropped to ~34.8" before the valve was fully closed.

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. Applicant may choose this if they believe that the level controller will close the valve at 33" (33" is the point at which the valve would open after it had closed on a lowering level of 30").

C is incorrect. 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 33 inches, the controller high output limit is returned to 100 percent and the controller opens the valve as required by the level error signal. Due to the very slowly lowering water level, the valve controller will close the letdown valve before a significant amount of inventory loss has occurred, level will stabilize at approximately 35", confirmed with simulator modeling.

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 approximately 35" RPV water level. Water level will not reach the RWCU system isolation setpoint of -38" RPV water level. Applicant may choose this if they believe level will continue down until the PCIS isolation of RWCU.

K/A: 204000 Reactor Water Cleanup System A4.08 3.4/3.4
K/A Statement: Ability to manually operate and/or monitor in the control room: reactor water level

References: TM-OP-061-ST Rev. 9
OP-161-001 Rev 50

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

A LOCA occurred in the U-1 Drywell, and::

- RPV level is +20 inches and is steady.
- Suppression Chamber pressure is 7.3 psig and rising slowly.
- The SRO 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, HV-151-F017B must be _(1)_ and Suppression Chamber Spray valves HV-151-F027B and HV-151-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 2 performing a startup

- HPCI and RCIC surveillances have just been completed
- The 'A' Loop of RHR is in Suppression Pool Cooling using 1A 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 'A' EDG is _____(1)______ and 'C' RHR pump _______(2)________ be used to restore suppression pool cooling from the Control Room.

A. (1) running
(2) can

B. (1) running
(2) can NOT

C. (1) in standby
(2) can

D. (1) in standby
(2) can NOT

Answer:

A is incorrect. 'A' EDG will start due to undervoltage on its bus and all breakers supplying the bus being open. It will be unable to close its output breaker to power the bus due to the differential overcurrent lockout of the bus. 'C' RHR pump cannot be used to restore suppression pool cooling due to RHR loop draindown and loss of power to RHR loop MOVs. Applicant may choose this answer if they do not recall the power lost to RHR loop MOVs required to restore the loop to operation.

B is correct. 'A' EDG will start due to undervoltage on its bus and all breakers supplying the bus being open. It will be unable to close its output breaker to power the bus due to the differential overcurrent lockout of the bus. 'C' RHR pump cannot be used to restore suppression pool cooling due to RHR loop draindown caused by trip of the 'A' RHR pump and open MOVs. This requires performance of a slow fill and vent of the RHR loop which cannot be accomplished due to loss of power to RHR loop MOVs.

C is incorrect. 'A' EDG does not remain in standby and will start due to bus undervoltage and all breakers supplying the bus being open. 'C' RHR pump cannot be used to restore suppression pool cooling due to RHR loop draindown and loss of power to RHR loop MOVs. Applicant may choose this answer if they do not recall the power lost to RHR loop MOVs required to restore the loop to operation, and believe that the bus lockout will cause the EDG to remain in standby.

D is incorrect. 'A' EDG does not remain in standby and will start due to bus undervoltage and all breakers supplying the bus being open. 'C' RHR pump cannot be used to restore suppression pool cooling due to RHR loop draindown and loss of power to RHR loop MOVs. Applicant may choose this answer if they recall the power lost to RHR loop MOVs, and believe that the bus lockout will cause the EDG to remain in standby.

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
AR-016-001 Rev. 44
TM-OP-024-ST Rev. 10

Applicant Ref: None

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

Question source: Modified Peach Bottom 12/08

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

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. 39 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?

	Unit 1	Unit 2
A.	Condensate Transfer	Condensate Transfer
B.	Demineralized Water	Condensate Transfer
C.	Condensate Transfer	Demineralized Water
D.	Demineralized Water	Demineralized Water

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   Fuel Pool Cooling and Clean-Up   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. 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 Engine Driven Fire Pump will start.

C. The Motor Driven and Diesel Engine Driven Fire Pumps will start.

D. Neither the Motor Driven nor Diesel Engine 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. **Incorrect:** 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: 2295B and 2301B, 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’, and ‘C’ reactor feed pumps are in service

B. ‘B’ and ‘C’ reactor feed pumps are in service, ‘A’ reactor feed pump is tripped

C. ‘A’ and ‘B’ reactor feed pumps are in service, ‘C’ reactor feed pump is tripped

D. ’A’ reactor feed pump is in service, ‘B’ and ‘C’ reactor feed pumps are 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?

	Power Supply	RSP Control
A.	4 kV ESS 1B	NO
B.	4 kV ESS 1B	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. ESS Bus 1B powers 2B RHRSW pump.

B. **Incorrect but plausible:** Plausible if the applicant does not recall that the RHRSW Pump 1B is powered from ESS 1D. ESS Bus 1B powers 2B RHRSW pump.

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: TM-OP-016, Rev 8 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

If the SGTS Makeup Outside Air Dampers FD-07551A2/B2 fail to close during a Secondary Containment Isolation (Zone 3) initiation signal, the excess airflow:

A. through the Makeup Outside Air Dampers will result in lower available airflow to the Cooling Outside Air Dampers, resulting in potential over temperature and fire in the High Efficiency Charcoal Adsorber Filter

B. may result in excessive current draw and overcurrent trip of the SGTS fan(s) which could in turn cause an unmonitored release due to a lack of negative pressure in the secondary containment

C. 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. will result in the inability of the SGTS to draw down the Secondary Containment within the required time and higher offsite post-LOCA doses

Answer:

A is incorrect. Over temperature 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 should not result in a fan overcurrent condition.
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

With Unit 1 initially operating at 100% power, 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 1.11

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. -161 inches level indicated is the safety limit. Reactor water level is required to be above the top of the active fuel (-161") to provide core cooling capability when the reactor is shutdown.

C. **Incorrect but plausible:** Reactor pressure rising to 1320 psig does not exceed the Safety Limit, which is 1325 psig, but is plausible since it is above the highest Safety valve setpoint

D. **Incorrect but plausible:** Incorrect Plausible since MCPR is below the SL for single loop operations.

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.

	Drywell Pressure (psig)	Suppression Chamber Pressure (psig)	Drywell Temp (°F)
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 Applicant Ref: None
TM-OP-059-FS Rev. 0

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 2.5 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)
- 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. Administrative Dose Control limits without a dose extension, ONLY

B. Administrative Dose Control limits with a valid dose extension, ONLY

C. BOTH Administrative Dose Control limits (with and without a valid dose extension).

D. BOTH Administrative Dose Control limits 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. **Correct:** 2500 mr/h x .75 = 1875 mr/hr + 1850 mr/hr = 3725 mr/hr. Result of 3725 mr/hr is above the Administrative Dose control limit of 2000 mr/hr, but below the Administrative Dose Control limits with a valid dose extension (4000 mr/hr), and NRC exposure limit of 5000 mr/hr.

B. **Incorrect but plausible:** Plausible if the applicant does not recall the Administrative Dose Control limits with a valid dose extension of 4000 mr/hr.

C. **Incorrect but plausible:** Plausible if the applicant does not recall the Administrative Dose Control limits with a valid dose extension of 4000 mr/hr.

D. **Incorrect but plausible:** Plausible if the applicant does not recall Adminstrative Dose control limits of 2000 mr/hr/4000 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 that required Operator B to hold a ladder for 2 of the 10 tags on the clearance. The System Operating Representative is ready to have the Clearance removal Independently Verified (IV).

To independently verify the Clearance removal, the System Operating Representative may use:

A. Operator B, as long as the verification is performed on a different day

B. Operator B for all 10 components

C. Operator B for the 8 tags he did not assist with and Operator C for the remaining 2

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 during positioning of two components, not eligible to be independent verifier per NDAP-QA-0027 for those two components, but not all ten.

C is correct. Operator B can verify eight of the tags, but cannot verify the two tags for which he/she was holding the ladder. Another operator must be used for these two tags.

D is incorrect. Operator A cannot be his/her own verifier, but Operator B can verify those tags which he/she did not assist with or reposition.

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 Applicant Ref: None
NDAP-QA-0027 Rev. 10

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

During a Station Blackout, Unit 1 is required per EO-100-030, “Unit 1 Response to Station Blackout,” to open 250 VDC load center breakers to 1D155 and 1D165. EO-200-030, “Unit 2 Response to Station Blackout,” does NOT direct the same actions for 2D155 and 2D165. Why **NOT**?

A. The 250 VDC Vital Batteries on Unit 2 have more storage capacity than their associated batteries on Unit 1.

B. The Non-Vital Loads that are shed on Unit 1 have a separate Non-Vital Battery Bank on Unit 2.

C. The Blue Max installation prevents this from being a problem on Unit 2.

D. Unit 2 has less Vital Loads on 250 VDC Batteries than Unit 1.

K&A Rating: 2.2.3 (3.8/3.9)

K&A Statement: **Knowledge of the design, procedural, and operational differences between units.**

Justification:

A. **Incorrect but plausible:** Plausible if the applicant does not understand the storage capacity of vital batteries, and does not recall that Unit 2 has a separate non-1E battery, 2D140, to ensure four hour capacity.

B. **Correct:** Unit 2 has a separate non-1E battery, 2D140, which carries non-1E MCC loads such as 2D155 and 2D165, to ensure four hour capacity.

C. **Incorrect but plausible:** Plausible if the applicant does not understand the purpose of blue max installation, which ensures long term vital loads operation beyond station blackout 4 hours coping time.

D. **Incorrect but plausible:** Plausible if the applicant does not understand the difference between Unit 1 and Unit 2 vital loads, and does not recall that the Unit 2 has a separate non-1E battery, 2D140, which carries loads such as 2D155 and 2D165.

References: N/A Applicant Ref: NONE

Learning Objective: N/A

Question source: SQ Bank Modified

Question History: Not used on 2011 or 2008 SQ Written

Cognitive level: Memory/Fundamental knowledge: X

10CFR 41.5 / 41.6 / 41.7 / 45.10 / 45.12

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 light extinguishes once the condition clears and 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 extinguishes 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. Both lights extinguish once their initiating condition clears and 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. Both lights automatically extinguish when their 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. Both alarms 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, both alarms do not automatically reset and must be manually reset at the Indicator and Trip Unit

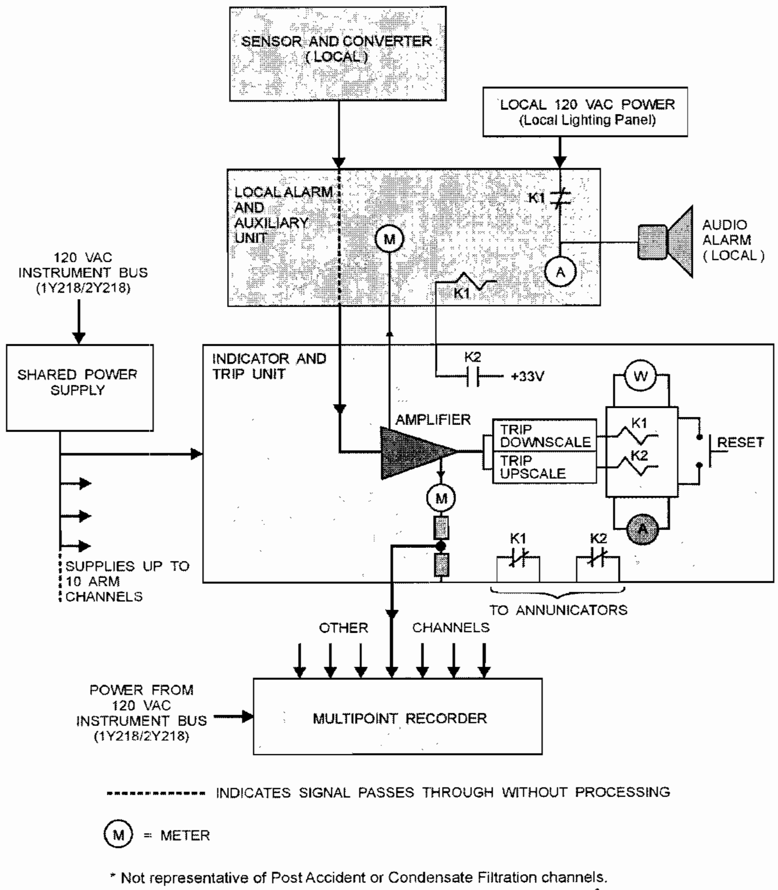
LOCAL ALARM AND AUXILIARY UNITS

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.



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: Not used on 2008 or 2011 Susquehanna exams

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 (pg 1 of 2)

Unit 1 is operating at 100% power. Spent fuel pool cask loading evolution is ongoing. An operator reports 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 and rising
- Fuel Pool Pp Area (Hi Alarm) radiation reading 10 R/Hr and rising
- All other reactor building radiation levels are below alarm setpoint.

REACTOR BUILDING RADIATION

ARM NUMBER				MAX NORMAL RADIATION	MAX SAFE RADIATION	
RB AREA EL (FT)	LO RANGE	HIGH RANGE	ARM CHANNEL DESCRIPTION	FIELD	FIELD (MR/HR)	EO104 (R/HR)
818	35+	N/A	CASK STOR AREA	**HI ALARM**	**10^4**	**10**
	14*	N/A	SPENT FUEL CRIT MON			
	15+	N/A	REFUEL FLOOR NORTH (SOUTH U-2)			
	42+	N/A	REFUEL FLOOR WEST			
	47* (44* U-2)	N/A	SPENT FUEL CRIT MON			
	N/A	49	REFUEL FLOOR AREA			
749	8+	52	RWCU RECIRC PP ACC	**HI ALARM**	**10^4**	**10**
	10*	54	FUEL POOL PP AREA			
	11+	N/A	RX BLD SAMPLE ST			
719	5*	50	CRD NORTH	**HI ALARM**	**10^4**	**10**
	6*	51	CRD SOUTH			
670	16+	53	REM SHDN ROOM ACC	**HI ALARM**	**10^4**	**10**
645	3+	48	HPCI PP*TURB ROOM	**HI ALARM**	**10^4**	**10**
	2+	57	RCIC PP*TURB ROOM			
	25*	55	RHR A*C PP ROOM			
	1*	56	RHR B*D PP ROOM			
	4*	N/A	RB/RW SUMP AREA			

Question 72 continued (pg 2 of 2)

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"
- Scram Reactor

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-101-001 (B17) RX WATER HI – LOW 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). In addition, it could be expected to receive AR-101-001 (B17) Rx Water Hi-Low Level due to the trip of the Condensate Pump and runback. If the operator confuses A17 with B17, they may choose this distracter.

C is correct. The above listed indications (alarm AR-101-001 A17) indicates that RPV level has exceeded the 54" trip setpoint for the RFP Turbines and Main Turbine, which should have caused a scram above 26% power. As such, the operator needs to place the Mode Switch in shutdown.

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)___ back to Bus 1D652.

A. (1) automatically transferring to alternate supply 1B236
(2) automatically transfers

B. (1) automatically transferring to alternate supply 1B246
(2) automatically transfers

C. (1) automatically transferring to alternate supply 1B236
(2) requires manual transfer

D. (1) automatically transferring to alternate supply 1B246
(2) requires manual transfer

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 MCC 1B236 not MCC 1B246 to maintain an uninterruptible power supply. MCC 1B246 provides alternate power source for Vital AC UPS 1D666.

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 MCC 1B246 provides alternate power source for Vital AC UPS 1D666, and 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, stable
- Seal #2 pressure is 10 psig, stable
- SEAL STAGE HI/LO FLOW annunciator is alarming
- SEAL LEAKAGE HI FLOW annunciator is 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 incorrect. 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 correct. If seal #2 failed, seal #2 pressure would drop. Seal leakage would rise causing the SEAL STAGE FLOW HI/LOW FLOW annunciator to alarm at 0.1 gpm and SEAL LEAKAGE HI FLOW annunciator will alarm.

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.

A transient occurs resulting in the following conditions:

- Scoop tube for Recirc MG Set B locks up
- “A” RRP runback to 48 percent occurs
- Loop "A" total Jet Pump flow is 24.9 million lbm/hr.
- Loop "B" total Jet Pump flow is 46.6 million lbm/hr.
- Total indicated core flow 71.5 million lbm/hr
- Core Plate DP on PICSY is 6.9 psid.

Two hours have elapsed since the transient, MCPR reached 1.11 shortly after the transient. Which of the following is(are) 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 41.6 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 the loop with lower flow to be declared "not in operation". 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.

<u>SRO Only Justification</u>: 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 TS 2.1.1	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

10CFR55 43(b)(2)

Comments:

Question 77

Unit 1 and 2 are at rated power. 'B' EDG is out of service for overhaul and 'E' EDG is substituting. 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 2.2 amps
- All other cells tested were within allowable values

Assuming that the above battery conditions CANNOT be corrected, which one of the following describes the <u>latest</u> that the plant can transition to 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 3.5/3.9
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:		Applicant Ref:
	TM-OP-002-ST Rev. 5	
	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	43(b)(2)

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, steady.
- Condensate Pump A (1P102A) bearing temperature is 270° F, up slow
- Condensate Pump B (1P102B) bearing temperature is 240° F, up slow
- Condensate Pump C (1P102C) bearing temperature is 230° F, steady
- Condensate Pump D (1P102D) bearing temperature is 250° F, steady
- EHC Fluid Temp TI-10180 is 160° F, steady

At the present time with the given plant conditions, which of the following actions are required by ON-115-001, “Loss of TBCCW” based on current values listed above?

A. Commence Plant Shutdown in accordance with GO-100-004, “Plant Shutdown to Minimum Power”.

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” ONLY.

D. Reduce Reactor Power IAW Reactor Engineering instructions in CRC book and remove Condensate Pump ‘A’ IAW OP-144-001, “Condensate System”.

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 two or more of the condensate pump's bearing temperatures were to exceed 265 F, then the answer would be correct to Scram, trip RFPs, 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 pigs.

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

10CFR55 43(b)(5)

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, select the necessary corrective action in accordance with EO-100-113, "Level/Power Control".

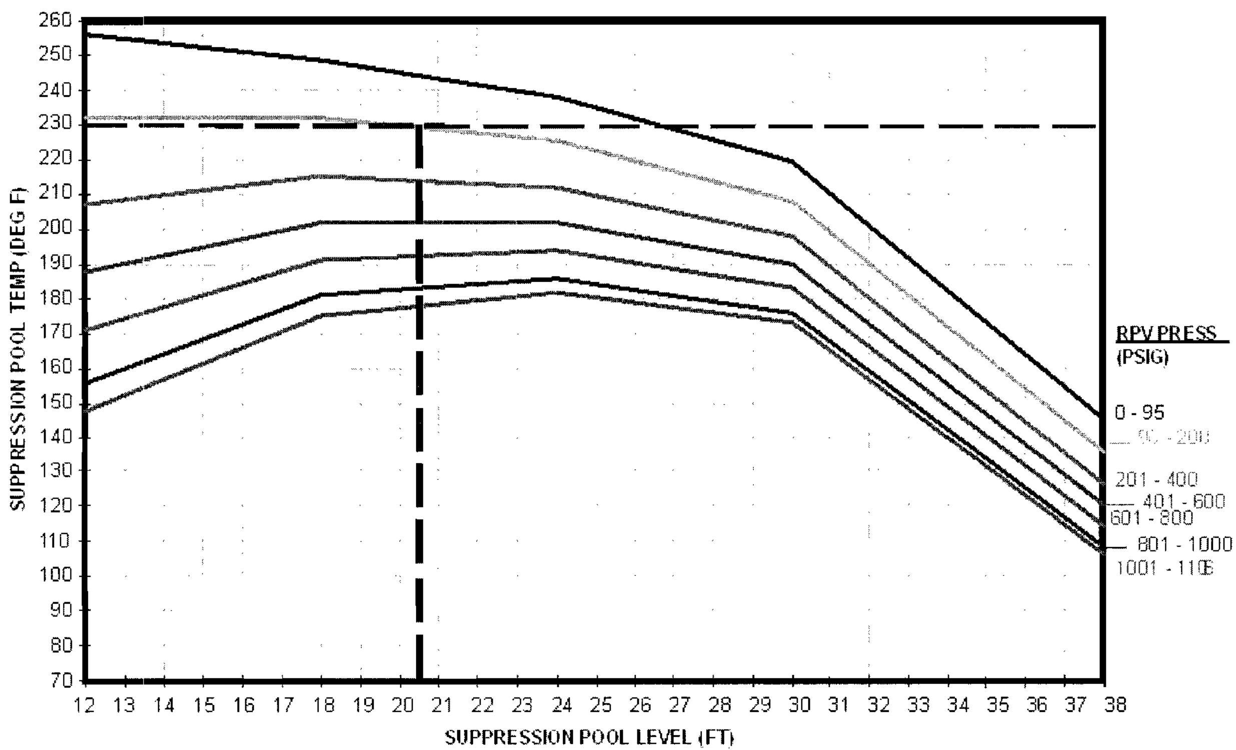
A. Perform EO-100-112, Rapid Depressurization

B. Anticipate Rapid Depressurization by opening all BPV irrespective of cooldown rate

C. Use BPVs to maintain RPV pressure below HCTL, not to exceed 100° F/hr

D. Use BPVs to maintain RPV pressure below HCTL, exceeding 100° F/hr if necessary

FIG 2 HCTL
HEAT CAPACITY TEMPERATURE LIMIT



Answer:

A is incorrect. While Rapid Depressurization is permissible with an ATWS in progress if initial ATWS power <5%, HCTL has not yet been exceeded.

B is incorrect. RPV Pressure reduction is permitted per LQ/P-4 to maintain RPV Pressure below HCTL. Use of bypass valves to rapidly depressurize the RPV if a Rapid Depressurization is anticipated is not permitted while in EO-100-113 Power/Level Control. This is permitted in EO-100-102 RPV control (RC/P-3), however, EO-100-102 RPV control is exited once EO-100-113 is entered for the ATWS. The "BPV" symbol accompanies step SP/T-8 in EO-100-103 and is <u>not</u> an instruction; it is attached to step SP/T-8 as a visual *reminder* of an existing instruction at RC/P-3. If the RPV Control procedure is not in use, <u>no</u> permission is given to use the bypass valves. Note, the bypass valve override is <u>not</u> located in EO-000-113, Level/Power Control. Applicants may confuse this and mistakenly believe that this authorizes them to anticipate Rapid Depressurization and open all BPV with an ATWS in progress.

C is incorrect. HCTL is not currently exceeded, but the limit is being approached. RPV Pressure reduction is permissible during an ATWS to prevent exceeding HCTL if the initial ATWS power <5%; exceeding cooldown rate limits is permitted.

D is correct. HCTL is not currently exceeded, but the limit is being approached. IAW EO-000-113 step LQ/P-4, RPV Pressure reduction is permissible during an ATWS to prevent exceeding HCTL if the initial ATWS power <5%; exceeding cooldown rate limits is permitted.

From EO-100-103:

SP/T-5 WHEN INITIAL ATWS PWR ≤5%

<u>OR</u>

RX SHUTDOWN WITH CONTROL RODS

CONTINUE

SP/T-8 WHEN RPV PRESS

SUPP POOL TEMP AND

SUPP POOL LVL

CANNOT BE MAINTAINED BELOW

FIG 2 HCTL

RAPID DEPRESS IS REQ'D

<u>EO-100-103 PC Control SP/T-8 bases</u>: 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, “HCTL,” is reached. Permission to use Main Turbine Bypass valves is given in an override located at RC/P-3 of RPV Control. This override allows the TS cooldown limit of 100°F/hr. to be exceeded. **The “BPV” symbol is <u>not</u> an instruction; it is attached to step SP/T-8 as a visual *reminder* of an existing instruction at RC/P-3. If the RPV Control procedure is not in use, <u>no</u> permission is given to use the bypass valves. Note, the bypass valve override is <u>not</u> located in EO-000-113, Level/Power Control**

K/A: 295026 Suppression Pool High Water Temp. G 2.1.25 3.9/4.2
K/A Statement: Ability to interpret reference materials, such as graphs, curves, tables, etc.

<u>SRO Only Justification</u>: 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 overall sequence of events/mitigative strategy of a procedure

References: EO-100-103 Rev. 13 Applicant Ref: None
EO-000-103 Bases Rev. 8
EO-100-113 Bases Rev. 8
EO-100-102 Bases Rev. 9

Learning Objective: PP002: 14618

Question source: New

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

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

10CFR55 43(b)(5)

Comments:

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, are 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
- 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 steps are required and what is the basis for the required actions?

	Required Action	**Basis**
A.	Rapid Depressurization	Downcomer openings are actually becoming uncovered and the ability to quench a LOCA is definitely threatened
B.	Open all BPVs	Rapid Depressurization is anticipated
C.	Trip RCIC and continue injecting with HPCI	HPCI is needed for adequate core cooling
D.	Trip and isolate HPCI	To prevent its 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

10CFR55 43(b)(5)

Comments:

Question 81

In accordance with EO-100-113, “Level/Power Control,” under what conditions can Core Spray be used to inject to the RPV during an ATWS?

A. Restoring level between +13” and +54” following injection of Hot Shutdown Boron Weight

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 bases, when Hot Shutdown Boron Weight is injected, level may be restored and maintained between +13" and +54", using the systems which were effectively providing RPV in the previous steps of the EO-113 flowchart. This would be the 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 43(b)(5)

Comments:

Question 82

At 1530 an unisolable steam leak occurred in the Turbine Building. The blowout panels opened and OSCAR is evaluating the release. At 1550 the combined Noble Gas SPING data is indicated to be 2.3 E8 μCi/min, and steady.

Reactor Conditions are as follows:

- All Rods in
- Containment Rad reading is 5 R/hr, and steady
- "A" MSL CANNOT be isolated.
- RPV Pressure is 800 psig, down slow.
- RPV Level is 40 inches, steady.
- RCIC is injecting.

In accordance with EO-100-102, "RPV Control," and E0-100-105, "Radioactivity Release Control," 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 less than 100° F /hr

C. Maintain RPV pressure between 800 to 1,050 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 E9 μCi/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°F /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°F /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 (Reference: PSTG RR-1 and EO-000-105, RR-5). RPV control does not direct operator to maintain pressure between 800 to 1,050 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 E9 μCi/min, and anticipated Rapid Depressurization actions to open BPVs is performed prior to reaching GE criteria.

<u>SRO Only Justification</u>: 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

Given the following:

- Unit 1 is operating at 100%
- 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

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 distracter for those candidates that recognize the Emergency classification condition and then select the reporting requirement to the State and County 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" 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	43(b)(5)	
Comments:		

Question 84

During a plant transient, the PCO observes actual RPV Level drop to -40 inches, before it turned and it is now + 10 inches and rising.

Assuming **NO** operator actions have been taken, at T= 15 minutes which of the following TS/TRM actions are required to be taken?

1. Obtain in-line conductivity measurement or grab sample once per 4 hours.
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.
3. Restore the Containment H2 and O2 Analyzers required channel(s) to OPERABLE status within 30 days
4. Restore secondary containment to OPERABLE status within 4 hours

A. 1 & 2 ONLY

B. 3 & 4 ONLY

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. **Correct:** Primary containment gas analyzers (CRM) isolate requiring primary containment atmospheric monitoring system to be operable and conductivity recorder is inoperable due to the containment isolation. Also, Containment H2 and O2 analyzers are isolated during containment isolation signals, however, they are auto bypassed after 10 minutes so the analyzers can be placed in service post-LOCA.

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 SBGT system.

C. **Incorrect but plausible:** Plausible if the applicant does not understand that Containment H2 and O2 analyzers are isolated during containment isolation signals, however, they are auto bypassed after 10 minutes. Primary containment gas analyzers (CRM) isolate requiring primary containment atmospheric monitoring system to be operable and conductivity recorder is inoperable due to the containment isolation.

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 SBGT 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:		Applicant Ref:
	ON-159-002, Rev. 29	
	TRM 3.4.1	TRM 3.4.1
	TS 3.4.6	TS 3.4.6
	TRM 3.3.4	TRM 3.3.4
	TS 3.6.4.1	TS 3.6.4.1

Learning Objective:

Question source:	New.	
Question History:		
Cognitive level:	Memory/Fundamental knowledge Comprehension/Analysis:	 X
10CFR55	43(b)(2)	
Comments:		

Question 85

Unit 1 is operating at

~~A Reactor Scram has occurred from~~ 100% power, when the following alarms and indications are received:

RCIC received a spurious actuation signal

- ~~RPV water level reached -40" on the transient, and is currently -18" up slow using feedwater~~
- ~~HPCI was placed on min flow~~
- RCIC flow controller output failed to minimum.
- RCIC Pump suction pressure upscale
- BOTH RCIC Equipment Room Riley indicators in ALARM and rising fast
- BOTH RCIC Emergency Cooler Riley indicators in ALARM and rising fast
- SIMPLEX Panel audible alarm
- AR-036-001 (B01), PUMP IS OPERATING
- AR-016-001 (G15), FIRE PROTECTION PANEL OC650 SYSTEM TROUBLE

(1) What is the source of the leak?
(2) What action must be taken to isolate the leak?
(3) What action is required in accordance with EO-100-104, "Secondary Containment Control"?

A. (1) Feedwater is discharging from a break at RCIC suction
(2) Close RCIC CST Suction Valve HV-149-F010
(3) Shutdown reactor in accordance with GO-100-004, Plant Shutdown to Minimum Power

B. (1) Feedwater is discharging from a break at RCIC suction
(2) Depress RCIC Trip Pushbutton
(3) Plant operation may continue until shutdown required by Tech Spec

C. (1) Break in steam line discharging into RCIC room
(2) Depress RCIC Isolation Pushbutton
(3) Shutdown reactor in accordance with GO-100-004, Plant Shutdown to Minimum Power

D. (1) Break in steam line discharging into RCIC room
(2) Manually close RCIC OB Steam Supply HV-149-F008
(3) Scram reactor and enter EO-100-102, RPV Control

Approved per telecom 1/23/12

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. **Incorrect but plausible:** A feedwater line break has occurred in the RCIC room. This is also confirmed by the upscale RCIC pump suction indication. 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. Closing the CST suction valve will isolate the suction from the CST, but will not isolate the leak. RCIC must be tripped to close/allow closure of the F013 valve. Applicant may choose this operation if they do not understand that the initiation signal keeps the F013 valve open, and if they believe that multiple areas are above max safe due to different sets of Riley indicators in alarm.

B. **Correct:** A feedwater line break has occurred in the RCIC room. This is also confirmed by the upscale RCIC pump suction indication. 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. In this case, RCIC must be tripped to close/allow closure of the F013 valve. No actions required per EO-100-104, plant shutdown will be forced by TS 3.5.3 after 14 days.

C. **Incorrect but plausible:** A feedwater line break has occurred in the RCIC room. Applicants may incorrectly believe that a steam leak is in progress due to elevated room temperatures if they do not also notice that the RCIC pump suction pressure is upscale. While isolating RCIC will provide a turbine trip and result in F013 valve closure, the action taken is not due to a steam leak. Applicant may also choose this answer if they believe that multiple areas are above max safe due to different sets of Riley indicators in alarm.

D. **Incorrect but plausible:** A feedwater line break has occurred in the RCIC room. Applicants may incorrectly believe that a steam leak is in progress due to elevated room temperatures if they do not also notice that the RCIC pump suction pressure is upscale. While manually closing the OB steam supply valve HV-149-F008 will serve to isolate a steam leak, there is no steam leak in progress to isolate. Applicant may also choose this answer if they believe a primary system is discharging into the secondary containment. In this case, feedwater is issuing from the leak, and the decision to scram is not until after leak isolation is attempted.

SRO Only Justification: This question is SRO only as it cannot be answered solely by knowing systems knowledge, immediate operator actions, entry conditions for EOP/AOP, or overall mitigative strategy of procedures. This question requires assessment of plant conditions, and selection of a procedure or section of a procedure to mitigate, recover, or with which to proceed.

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
10CFR55	43(b)(5)	
Comments:		

Question 86

Unit 1 was at 100% when the following conditions occur:

- 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 35.5 feet and rising slowly.

In accordance with EO-100-103, "PC Control", which one of the following steps would be correct and why?

A. Open ALL BPVs because drywell vacuum breakers are beginning to become submerged.

B. Rapid depressurization because drywell vacuum breakers are beginning to become submerged.

C. Open ALL BPVs to depressurize the RPV prior to exceeding the capability of the SRV tail pipe and tail pipe supports.

D. Rapid depressurization prior to exceeding the capability of the SRV tail pipe and tail pipe supports.

K&A Rating: 239002 G2.4.16 (4.4)

K&A Statement: **Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines.**

Justification:

A. **Incorrect but plausible:** Plausible, partially correct that if rapid depressurization is anticipated and the reactor is shutdown on control rods, then Main Turbine Bypass Valve may be used to depressurize the RPV until the rapid depressurization limit, "38 Feet" is reached. Note that 38 feet is not yet reached, however, the reason for depressurization is to prevent exceeding the capability of the SRV tail pipe, tail pipe supports, quencher, or quencher supports. At 43' the drywell vacuum breakers begin to cover.

B. **Incorrect but plausible:** Plausible if the applicant does not recall rapid depressurization is anticipated and the reactor is shutdown on control rods, then Main Turbine Bypass Valve may be used to depressurize the RPV until the rapid depressurization limit, "38 Feet" is reached. Note that 38 feet is not yet reached.

C. **Correct:** Rapid depressurization is anticipated and the reactor is shutdown on control rods, then Main Turbine Bypass Valve should be used to depressurize the RPV until the rapid depressurization limit, "38 Feet" is reached. Note that 38 feet is not yet reached, the reason for depressurization is to prevent exceeding the capability of the SRV tail pipe, tail pipe supports, quencher, or quencher supports.

   Note: HPCI and RCIC were placed in min flow because EO-100-103, requires HPCI and RCIC to be started prior to reaching 25', because if pool level were above 25', water could enter the exhaust piping and a high exhaust pressure would occur when the system is started due to inertial effect of water in the piping. This could potentially cause the system to trip on high turbine exhaust pressure. Therefore, both HPCI and RCIC are ensured to be running when pool level reaches 25'.

D. **Incorrect but plausible:** Plausible, partially correct that if the reason for depressurization is to prevent exceeding the capability of the SRV tail pipe, tail pipe supports, quencher, or quencher supports, however if rapid depressurization is anticipated and the reactor is shutdown on control rods, then Main Turbine Bypass Valve should be used to depressurize the RPV until the rapid depressurization limit, "38 Feet" is reached. Note that 38 feet is not yet reached.

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:
Comprehension/Analysis X

10CFR55 43(b)(5)

Comments:

Question 87

Unit 2 is experiencing an ATWS. All attempts so far to insert control rods and inject SLC have been unsuccessful. Plant conditions as follows:

- Reactor power is 14%
- Feedwater, condensate, and RCIC are unavailable
- Reactor water level is being maintained between -60" and -110" using HPCI
- Suppression Pool temperature is 149°F and rising slowly

Two minutes later, the following additional plant conditions exist:

- AR-214-001 (E01) Condensate Storage Tank LO Water Level
- AR-214-001 (F02) HPCI Pump Suction LO Press
- AR-214-001 (C02) HPCI Turbine Trip Solenoid Energized
- AR-214-001 (A01) HPCI Turbine Tripped

What procedure(s) are required to restore level control?

A. Prevent injection from LPCI/Core Spray, 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-214-001 (E01) and restore HPCI to service in accordance with OP-252-001.

Answer:
A is correct. With the suppression pool temperature >140°F, direction from EO-200-103 is to ensure HPCI suction aligned to the CST. With the CST unavailable, neither HPCI nor RCIC are available for high pressure injection. With RCIC previously stated as unavailable and with no high and medium pressure injection available, CRD and SLC are incapable of maintaining reactor water level at this reactor power. EO-200-113 LQ/L-14 directs going to LQ/L-18 (prevention of injection from FW/Condensate/LPCI/Core Spray), performing rapid depressurization and restoring level control using low pressure ECCS due to the inability to restore and maintain reactor water level >-161".

B is incorrect. HPCI cannot be recovered without first transferring suction to the Suppression Pool, which is not allowed by EO-100-103, step SP/T-4. This step directs transferring suction to CST, which is unavailable based upon low level alarm and HPCI tripping on low suction pressure. If HPCI suction were left on the Suppression Pool, HPCI oil cooler outlet would continue to rise until HPCI turbine bearings were damaged.

C is incorrect. Applicant may believe that HPCI turbine isolation has occurred based upon conditions given, although no isolation has occurred. Additionally, HPCI cannot be recovered without first transferring suction to the Suppression Pool, which is currently not allowed per EO-100-103, step SP/T-4.

D is incorrect. Automatic suction transfer to the Suppression Pool has failed. With the suppression pool temperature >140°F, direction from EO-200-103 is to ensure HPCI suction aligned to the CST. With the CST unavailable, HPCI is not available for high pressure injection.

K/A: 206000 High Pressure Coolant Injection System A2.08 3.9/4.2
K/A Statement: ability to (a) predict the impacts of high suppression pool temperature on the High Pressure Coolant Injection system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations
SRO Only Justification: This question is SRO only since SRO candidates must assess plant conditions and select the correct 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: 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: New

Question History:	None	
Cognitive level:	Memory/Fundamental knowledge:	
	Comprehension/Analysis:	X
10CFR55	43(b)(5)	

Comments:

New

Question 88

Unit 1 is operating at 95 percent power.

The following conditions exist:

- The following Zone 1 Recirculation dampers failed surveillance SO-134-003, "Zone 1 Recirc Isolation Damper Timing and were declared INOP:
    - HD17601A, Zone 1 Filt Exh to Recic System Damper
    - HD17602A, Zone 1 Exh System to Recirc System Damper
- A functional test of the Standby Gas Treatment (SGTS) Fire Suppression System DS-093 for 'A' SGTS Filter 0F169A and DS-094 for 'B' SGTS Filter 0F169B were also performed and test 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 3.6.4.3 B.3.6.4.3 TRM 3.7.3.2	Applicant Ref: TS 3.6.4.3

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

10CFR55	43(b)(2)

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 ensures adequate fuel oil for a minimum of....

A. 55 minutes of DG A-D operation at the maximum post accident load demand conditions

B. 55 minutes of DG A-D operation at DG continuous rated load conditions

C. 62 minutes of DG A-D operation at the maximum post accident load demand conditions

D. 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:	

10CFR55 43(b)(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 compressors are maintaining 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. In addition, per the 3.8.1 bases, T-10 is INOP if its tap changer is not in automatic so one offsite circuit is also INOP.

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
Comprehension/Analysis: X

10CFR55 43(b)(2)

Comments:

Question 91

Unit 1 refueling operations are in progress with the following plant conditions:

- Bridge movements are being controlled using the joysticks
- A bundle is being lifted from the core
- As the bundle clears the top guide, a momentary loss of power occurs to the Refuel Platform.

Based on the above, in accordance with ON-081-002, "Refueling Platform Operation Anomaly"....

The Fault Lockout ____(1)____ have to be physically reset in 1C253, and permission must be granted by the _____(2)_____ with concurrence of the applicable Unit Supervisor and Refuel Floor Manager, to resume fuel handling activities.

A. (1) will NOT
(2) Shift Manager

B. (1) will
(2) Shift Manager

C. (1) will NOT
(2) Assistant Ops Manager – Shift/WCC

D. (1) will
(2) Assistant Ops Manager – Shift/WCC

From TM-OP-081A-PG Fuel Handling Unit 1:

- When STOP is depressed, a fault lockout icon will appear on the CRT touch-screen.
- Touching this icon will bring up the fault menu and "Hoist Drive Faults" and "Main Power OFF (Max)" will be highlighted.
- These faults will auto clear when the START pushbutton is depressed

Main Power Off Contact (MAX) - Signifies that the stop button has been pushed. A contact from the Start Relay is fed back into the PLC to monitor the system power. If control power is lost to the machine, a fault is generated

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.

Once cause has been identified and corrected, ON-081-002 directs proceeding to step 4.15.

1.5 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 if electrical failure fault lockout automatically resets when START pushbutton is pressed, may choose this answer. Shift manager is a plausible choice for requesting permission to resume fuel handling activities, as the shift manager is responsible for all shift operations.

B. **Incorrect but plausible:** If candidate does not recall if electrical failure fault lockout automatically resets when START pushbutton is pressed, may choose this answer. 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 power is lost and subsequently returned to the refuel platform, the platform generates a fault lockout due to loss of power. Unlike a fault lockout for the Main Hoist Motor Drive Faults, which require manually resetting in panel 1C253 Refuel Platform trolley cabinet, this fault clears when the START pushbutton is depressed. 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:** The fault lockout in this case is reset by pushing the START pushbutton, and does not require to be physically reset. Shift Manager permission is insufficient to resume fuel handling activities. Permission from Assistant Ops Manager – Shift and concurrence from applicable Unit Supervisor and Refuel Floor Manager is required before resuming fuel handling activities .

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
10CFR55	43(b)(7)	

Comments:

Question 92

Which of the following is the minimum required to meet the TRO for a Fire Suppression Water Supply Subsystem?

A. Only the Clarified Water Storage Tank (0T523) with a minimum volume of 300,000 gallons. No other source will maintain the subsystem operable.

B. The Clarified Water Storage Tank (0T523) 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 (0T523) with a minimum volume of 300,000 gallons, AND EITHER the Unit 1 or Unit 2 Cooling Tower Basin

D. EITHER the Unit 1 Cooling Tower Basin OR Clarified Water Storage Tank (0T523) with a minimum volume of 300,000 gallons

K&A Rating: 286000 G2.2.25 Fire Protection (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 for fire suppression water operability, one water supply from either U1 or U2 cooling tower basin can be utilized besides Clarified Water Storage Tank (0T523) with a minimum volume of 300,000.

B. **Incorrect but plausible:** Plausible if the applicant does not recalls that only ONE source is required for operability, either U1 cooling tower basin, U2 cooling tower basin, OR Clarified Water Storage Tank (0T523) 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 (0T523) with a minimum volume of 300,000.

D. **Correct:** Only ONE source is required for operability, either U1 cooling tower basin, OR Clarified Water Storage Tank (0T523) with a minimum volume of 300,000 meets the TRO requirement.

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 3.7.3.1
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:

10CFR55 43(b)(2)

Comments:

Question 93

Unit 1 is in a refueling outage in Mode 5. There is **NO** fuel movement in progress. The 'A' CREOASS System is OPERABLE.

The 24 month surveillance for 'B' CREOASS 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 Unit Supervisor, 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 **WITHOUT** any additional evaluations, and,

(2) The basis for that decision?

	(1)	(2)
A.	Maintenance activity **CANNOT** be authorized	LCO 3.7.3 is currently **NOT** met and entry into OPDRV specified condition can **ONLY** be done when two CREOASS 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 CREOASS 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 an 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 in pressurization mode for an unlimited time, therefore the maintenance activity may be authorized.

B. **Correct:** because the 'B' CREOASS 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 allow continued operation for an unlimited time. IAW TS 3.7.3.D, with the operable CREOASS 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' CREOASS 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.

<u>SRO Only Justification</u>: 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

10CFR55 43(b)(2)

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 the inoperable equipment must be restored within 72 hours, or be in Mode 3 in 12 hours and Mode 4 in 36 hours.

At 0800 on 2/19, the surveillance is completed, and is determined to be UNSAT.

Which one 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 2000 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 2/24.

K&A Rating: 2.1.1 (4.2)

K&A Statement: **Knowledge of conduct of operations requirements.**

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 and additional 36 hours applies towards Cold shutdown requirements.

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. However, the surveillance was performed within the grace period, but determined to be UNSAT on 2/19 @ 0800. Therefore, SR 3.0.1 applies, and 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. SR 3.0.1 states that failure to meet a surveillance, whether such failure is experienced during the performance of the surveillance or between performances of the surveillance, shall be failure to meet the LCO.

D. **Incorrect but plausible:** Plausible if the applicant determines that cold shutdown is required within 48 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:
Comprehensive/Analysis: X

10CFR55 43(b)(2)

Question 95

Unit 1 is currently at 63% power with HPCI running for quarterly flow verification The following conditions now exist:

- A transient has occurred which has caused major fuel damage
- RCIC Equipment Area temperature is 79°F, steady
- HPCI Equipment Area temperature is 93°F, up slow
- HPCI Equipment Area ARM is reading 13.6 R/hour, up slow
- RCIC Equipment Area ARM is reading 11.3 R/hour, up slow
- CRD North and South reading 14.2 R/hour, up slow

Based upon the given conditions, and in accordance with EO-100-104, a primary system is.....

A. discharging into the Reactor Building, cooldown the reactor in accordance with EO-100-102, RPV Control

B. discharging into the Reactor Building, enter and perform EO-100-112, Rapid Depressurization

C. NOT discharging into the Reactor Building, Scram and depressurize the reactor in accordance with EO-100-102, RPV Control

D. NOT discharging into the Reactor Building, shutdown 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, based upon elevated HPCI room temperatures if they do not notice that HPCI is already running. 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, based upon elevated HPCI room temperatures and did not notice that HPCI was also operating. If applicant believed that a primary system were discharging into the secondary containment, two or more areas above max safe radiation level, they would choose 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 may be prudent to scram the reactor based upon radiation levels in the plant, but it is not the procedurally directed action per EO-100-104

D. **Correct:** A primary system is NOT discharging into the Reactor Building; depressurization in accordance with GO-100-004, Plant Shutdown to Minimum Power is required. This is due to the fact that a primary system is not discharging into the secondary containment, and two areas are above max safe radiation levels.

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
10CFR55	43(b)(5)	

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 the 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 1B120 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 1B120 to determine any required compensatory actions.

C. None. 1B120 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 1B120 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 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

10CFR55 43(b)(5)

Comments:

Question 97

Which one of the following switching orders requires a 10CFR50.59 review per OI-AD-085, Operations Switching Orders?

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 Reactor Building Zone III Exhaust Dampers HD-17538A and HD-17538B 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:

5.1.1 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

5.2.3 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:	AD044: 14891	
Question source:	SSES bank 131	
Question History:	Not used on SSES 2008 or 2011 written exams	
Cognitive level:	Memory/Fundamental knowledge Comprehension/Analysis:	 X
10CFR55	43(b)(3)	

Comments:

SSES 2011 exam question 95 covers similar topic but different question

Question 98

With a liquid effluent discharge of a set of Sample Tanks is in progress the 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 ONLY to lower the concentration through additional dilution.

C. Start an additional River Water Makeup Pump and open the blowdown flow control valve to lower the concentration through dilution.

D. Ensure the LRW Sample Tank Pump for the tanks being released automatically 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 and 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:

10CFR55 43(b)(5)

Comments:

Question 99

A refueling outage is in progress. The following conditions exist:

- Reactor cavity is flooded
- Reactor cavity and cask storage pit gates are removed

In accordance with NDAP-QA-0626, "Radiologically Controlled Area Access and Radiation Work Permit (RWP) System," which one of the following activities requires 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. Replacing a Jet Pump hold down beam

From NDAP-QA-0626:

Access control requirements for upper elevations of the Drywell, during outages

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 <u>provided that none of the following activities is being performed</u>:

- 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:** NDAP-QA-0626 restricts access to the upper drywell elevations for Jet Pump Mixing section movement, but not for Jet Pump holddown beam replacement.

<u>SRO Only Justification</u>: 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 Comprehension/Analysis:	X
10CFR55	43(b)(4)	
Comments:		

Question 100

The plant is operating at 100% power with the following:

Time (hh:mm)	Event
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?

	Latest Time For Declaration	Level of Emergency Declaration
A.	00:25	Declare and report the UNUSUAL EVENT
B.	00:32	Declare and report the UNUSUAL EVENT
C.	00:25	Declare and report the ALERT
D.	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. Since only 15 minutes is allowed to make the classification, the 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

10CFR55 43(b)(1)

Comments: