DRAFT

EXAMINATION ANSWER KEY

NRC 2004 RO WRITTEN EXAMINATION

1 **SYSID: 21001** **Points: 1.00**

The plant is at 100% power when Reactor Recirculation Pump (RRP) 14 develops a NON-CATASTROPHIC seal leak from both the high-pressure and low-pressure seals. The RO performs the following actions:

- Closes REACTOR R PUMP 14 DISCHARGE VALVE and then applies a momentary (2 second) OPEN signal
- Trips RRP 14
- Closes REACTOR R PUMP 14 SUCTION VALVE and then applies a momentary (2 second) OPEN signal.
- Closes REACTOR R PUMP 14 BYPASS VALVE.
- NO other operator actions are taken.

Which one of the following is the REACTOR POWER LIMIT and and the reason for this limit upon completion of these actions?

A. 90.5% because actions necessary to preclude an inadvertent start of RRP 14 without being warmed are not taken.

B. 100% because actions necessary to preclude an inadvertent start of RRP 14 with a cold leg are achieved when the pump is isolated.

C. 90.5% because the APRM flow-biased scram and rod block trip set points are non-conservative until the suction and discharge valves are re-closed.

D. 100% because the APRM flow-biased scram and rod block trip set points are restored to operable status when the discharge valve is initially closed.

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

NRC 2004 RO WRITTEN EXAMINATION

Question 1 Details

Question Type:	Multiple Choice
Topic:	NRC RO 1
System ID:	21001
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-202-1-01, EO-1.6

Per N1-OP-1, H.3.0 ~~D.4.0~~, ~~Reactor Power must be less than 90.5% prior to isolating a Reactor Recirculation Pump.~~

When operating with four recirculation loops in operation and one loop isolated, reactor power shall be limited to 90.5% unless the following conditions are met to preclude inadvertent startup of a recirculation pump with a cold leg:

1. The suction, discharge, and discharge bypass valves in isolated loop are fully closed and associated motor breakers locked open AND
2. Associated pump motor circuit breaker is open and breaker removed.

Non-conservative Recirc Flowbiased APRM scram and rod block trip setpoints due to the reverse-flow through the non-isolated Recirc Loop still being measured as part of total core flow. The Recirc Flow-biased APRM scram and rod block trip functions are inoperable until the tripped Recirc Pump's associated discharge OR suction valve is "closed". The tripped loop's discharge OR suction valve is still considered "closed" even when it is given an "open" signal for 2-3 seconds for valve stem warm-up, since reverse-flow is negligible.
Following closure, RRP suction and discharge valves are given a 2 to 3 second open signal to provide for valve stem growth to prevent Limitorque lockup.

References Provided None

Question 1 Cross References (table item links)

10CFR55

- 41(b)(5)
- 41(b)(10)

Cognitive Level

- 2

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 295001 AK3.05 3.2/3.6 Reduced loop operating requirements: Plant-Specific. AK3.05 Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Reduced loop operating requirements.

LP

- O1-OPS-001-202-1-01 Rev. na

Question Source

- New

PROC

- N1-OP-1 Rev. NA

Question Setting

- C1 (License class closed reference)

2 **SYSID: 21002** **Points: 1.00**

The plant is at 100% power with no equipment out of service.

- A disturbance on the 115kv line results in a loss of 115kv
- Breakers R10 and R40 open
- Protective relays at Lighthouse Hill clear necessary busses
- Bennetts Bridge auto-transfers to energize the line from Bennetts Bridge to Lighthouse Hill, energizing line #4 to J. A. Fitzpatrick and to NMP1

With respect to the conditions above, which one of the following describes how 115 Kv power will be restored?

A. Breaker R-40 closes to restore 115 Kv power

B. Breaker R-10 closes to restore 115 Kv power

C. Disconnect 178 closes to restore 115 Kv power

D. Disconnect 8106 closes to restore 115 Kv power

Answer: A

Associated objective(s):

Development Area (FIO)

Question 2 Details

Question Type:	Multiple Choice
Topic:	NRC RO 2
System ID:	21002
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-262-1-01, EO-1.7

Answer:	a. is correct Per N1-OP-33A section B system description, R-40 closes following the disturbance
Distractor:	No power available for R-10
Distractor:	Faulted transformer disconnect SW 168 (101N) or SW 178 (101S) opens on transformer lockout.
Distractor:	IF reclosure fails, bus sectionalizing disconnect SW 8106 opens AND THEN R10 and R40 attempt another reclosure to re-energize the unfaulted section of the 115 kV bus

References Provided: None

Question 2 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 2

Difficulty Level

- *Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.*

NUREG 1123 KA Catalog Rev. 2

- 295003 AA2.04 3.5/3.7 System lineups
AA2.04 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: System lineups.

Question Source

- New

PROC

- N1-OP-33A Rev. NA

Question Setting

- C1 (License class closed reference)

3 **SYSID: 21003** **Points: 1.00**

Following a loss of all off-site power the following conditions exist:

- Off-site power has been restored to PB12 from the Station Service Transformer through Breaker R122 and Emergency DG 103 is shutdown.
- PB102 is being supplied by emergency DG 102, Breaker R113 is open
- PB167 is powered from its normal source
- Panel 167A is being supplied from I & C Bus 130.

You have been directed to transfer the power source for PB167A from I & C Bus 130 to PB167

Which one of the following conditions apply to this transfer?

A. This transfer **CANNOT** be performed at this time because the normal power supply is powering PB167.

B. The power supply to Panel 167A from I&C Bus 130 must be opened before the power supply from PB167 can be closed.

C. This transfer **CANNOT** be performed at this time because it would result in both Powerboards being supplied from the same source.

D. The PB13 Section C to PB13 Section B Tie Breaker must be closed before Breaker #1 from Transformer 167 can be closed.

Answer: B

Associated objective(s):

Development Area (FIO)

Question 3 Details

Question Type:	Multiple Choice
Topic:	NRC RO 3
System ID:	21003
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-262-1-02, EO-1.6

Answer: B. The power supply to PB167A from PB130 must be opened before the power supply from PB167 can be closed. Opening the power supply from PB130 would de-energize PB167A. This would permit the dead bus transfer as required by N1-OP-30 Precaution and Limitation #19. Understanding that PB167A must be de-energized (dead bus) explains and applies the precaution.

Distractor: This transfer per Abnormal section 52.0 of N1-OP-30 **CANNOT** be performed at this time because this would close a breaker across two out of phase systems. However a dead bus transfer may be performed.

Distractor: The transfer can be performed and it would be much easier if both PBs were supplied form the same source. This distracter is used because several procedural precautions exist about cross-tying PBs and losing divisional separation.

Distractor: Closing the PB13 Section C to PB 13 Section B Tie Breaker would result in attempting to close a breaker between two out of phase systems. Although this breaker lineup is physically possible it is not in any procedure and is prevented by N1-OP-30 Precaution and Limitation #19.

References Provided: None

Question 3 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 2

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- G2.1.32 3.4/3.8 Ability to explain and apply system limits and precautions
- 295003 Partial or Complete Loss of A.C. Power

Question Source

- New

PROC

- N1-OP-30 Rev. NA

Question Setting

- C1 (License class closed reference)

4 **SYSID: 21004** **Points: 1.00**

During a Station Blackout, which one of the following is the reason for performing designated Battery Load Reductions within thirty (30) minutes of the start of the station blackout?

A. Support a manual dead bus transfer to MG Set 167.

B. Support a manual dead bus transfer to the standby static battery chargers.

C. Maintain power to reactor instrumentation, EC controls, and to start an EDG.

D. Avoid a loss of critical battery board loads due to breaker trips on over current.

Answer: C

Associated objective(s):

Development Area (FIO)

Question 4 Details

Question Type:	Multiple Choice
Topic:	NRC RO 4
System ID:	21004
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-263-1-01, EO-1.8

Answer: c. Per OP-47A, B.1.0: With no AC power (loss of both Off-Site power feeds and failure of both Diesel Generators to start), stripping of certain 125 VDC loads is required to ensure sufficient DC capacity for Reactor Instruments, Emergency Condenser controls, and to start a Diesel Generator. Directions for which loads to strip and when is given in N1-SOP-18.

Distractor: b. Static battery chargers are not available until AC power is restored. The Static chargers provide all of the DC power required for normal station operation when AC power is available. When both SR Static chargers for a battery are out of service (one for Q battery) or lose AC power, its associated battery will supply the 125 VDC loads. Each pair of SR static chargers are powered from a separate 600V bus, capable of being fed from the Emergency Diesel Generators, if normal power is lost. With a SBO, diesel generators are not available.

Distractor: a. MG Set 167 (computer supply) may also be aligned to charge the batteries, but is non-safety related. This transfer is performed manually at the Motor Generator Set control cubicle. MG Set 167 has no power. It is a spare and can be used as an emergency power source but has no power. Also, MG 167 DC motor breakers are stripped within 2 hours of the SBO and result in the loss of Process Computer and

Annunciators (Alarm Bus).

Distractor: d. Breakers on the SR battery boards have been bypassed. Fuses have been installed in their place in the back of the Battery Boards. Both the Negative and Positive legs to each load are fused.

References Provided: None

Question 4 Cross References (table item links)

10CFR55
- 41(b)(5)

Cognitive Level
- 1

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and *determine the correct answer.*

NUREG 1123 KA Catalog Rev. 2
- 295004 AK3.01 2.6/3.1 Load shedding: Plant-Specific

AK3.01 Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER:
Load shedding.

Question Source
- Modified

PROC
- N1-OP-47A Rev. NA

Question Setting
- C1 (License class closed reference)

5 **SYSID: 21005** **Points: 1.00**

Several minutes after a main turbine trip and reactor scram from 98% power the following conditions exist:

- RPV water level is 79 inches and slowly rising.
- Feedwater flow is less than 1.9×10^6 lbm/hr on motor driven pumps.
- All primary containment parameters are normal.

To restore normal feedwater control which one of the following actions is required prior to depressing the FEEDWATER RETURN TO NORMAL AFTER HPCI pushbuttons?

A. Reset the reactor scram.

B. Reset the turbine trip logic.

C. Place the Reactor Mode Switch in SHUTDOWN.

D. Take manual control of 11 and 12 FWP Flow Control Valves.

Answer: D

Associated objective(s):

Development Area (FIO)

Question 5 Details

Question Type:	Multiple Choice
Topic:	NRC RO 5
System ID:	21005
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-259-1-01, EO-1.8

Answer:	d. N1-OP-16 directs the operator to take manual control of the FWP Flow Control Valves prior to restoring normal feedwater control. This prevents a FW Flow transient, which would return the FWP Flow Controls Valves to the HPCI mode.
Distractor:	a.There is no need to reset the reactor scram, RPS does not initiate HPCI.
Distractor:	b.There is no need to reset the turbine logic, although the turbine trip logic initiated HPCI, it is not required to reset HPCI.
Distractor:	c. Placing the REACTOR Mode Switch in SHUTDOWN does not effect HPCI.

References Provided: None

Question 5 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 2

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 295005 AK2.09 4/4.3 Feedwater - HPCI: BWR-2
AK2.09 Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Feedwater - HPCI: BWR-2.

Question Source

- New

PROC

- N1-OP-16, Rev. NA

Question Setting

- C1 (License class closed reference)

6 **SYSID: 21006** **Points: 1.00**

The plant is at 100% with Feed Water Pump (FWP) #11 and FWP #13 in service. A reactor scram occurs. The following conditions exist immediately after the scram:

- Reactor pressure is 950 psig and steady
- RPV level reached a low of +48 inches
- RPV level is currently +51 inches and is rising slowly
- FW LVL SP SETDN INIT light is OFF

Per SOP-1, Reactor Scram, which one of the following is the required operator action in response to these conditions?

A. Override FW LEVEL SETPOINT SETDOWN.

B. Manually initiate FW LEVEL SETPOINT SETDOWN.

C. Set FEEDWATER MASTER CONTROLLER at +55 inches.

D. Set FEEDWATER PUMP 11 M/A station at 50% in MANUAL.

Answer: C

Associated objective(s):

Development Area (FIO)

Question 6 Details

Question Type:	Multiple Choice
Topic:	NRC RO 6
System ID:	21006
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-342-1-01, EO-1.2

Answer: C ~~a~~. Setpoint set down failed. Per SOP-1 Immediate Operator Action (overrides) set the Master FW controller to +55 inches if FW LVL setpoint setdown fails to initiate (FW LVL SP SETDN INIT light is OFF and should be on; failed to initiate).

Immediately following a reactor scram sensed by a set of RPS Channel 11 and 12 scram relay contacts in RPS BUS 11 CKT 12, coincident with a reactor low water level signal of 52 inches, the K8 initiation relay will **automatically switch control of RPV water level from the master controller ID66 (ID15A) to level controller ID66B with a setpoint of 45 inches.** Relay K9 contacts will light a light on control console E. ID66B will track the output of ID15A (GEMAC M/A Station for FCV #13) to provide a bumpless transfer of the control signal. Setdown of the level setpoint will affect the operation of Feedwater Pump No. 13 Flow

Control Valve FCV-29-134 only. After water level recovers above the low level setpoint of 52 inches, the feedwater level Setdown circuit can be reset using Reset Switch on control console E, or placing 29-169, FW LVL SETPOINT SETDOWN to OVERRIDE, then back to NORMAL. Reactor water level control will return to the master Controller ID66.

An override switch located on Panel F can be also used to override the setdown signal at any time. This override restores normal control to ID66 allowing normal control of FCV #13 from ID15A.

Distractor: a. There is no benefit to override FW LVL setpoint setdown because it failed. Overriding at this time will not change the state of the circuit.

Distractor: b. Per SOP-1 Immediate Operator Action (overrides) set the Master FW controller to +55 inches if FW LVL setpoint setdown fails to initiate (FW LVL SP SETDN INIT light is OFF and should be on; failed to initiate). There is no benefit from initiating FW LVL setpoint setdown at this time. Action must be taken to minimize the RPV level rise to stay below the high level trip.

Distractor: d.When RPV level is above +53" (not +51") set FEEDWATER PUMP 11 M/A station at 0% (not 50%) in MANUAL to avoid overfeeding.

References Provided: None

Question 6 Cross References (table item links)

10CFR55
- 41(b)(7)

Cognitive Level
- 2

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- 295006 AA1.02 3.9/3.8 Reactor water level control system

AA1.02 Ability to operate and/or monitor the following as they apply to SCRAM: Reactor water level control system.

Question Source
- Bank

PROC
- N1-SOP-01 Rev. NA

Question Setting
- C1 (License class closed reference)

7 **SYSID: 21007** **Points: 1.00**

The following plant conditions exist:

- Reactor pressure is 1090 psig and lowering slowly
- Reactor water level is 100 inches and lowering slowly
- Both ECs have automatically initiated before establishing control at RSP 11
- One (1) minute later Channel 11 Control Transfer Switch is placed in EMER at RSP 11

Per SOP-9.1, Control Room Evacuation, which one of the following actions can be taken to control the cool down rate from RSP 11?

A. Place CHANNEL 11 CONTROL TRANSFER switch in EMER and then cycle EC Steam Supply IV (39-09R).

B. Place CHANNEL 11 CONTROL TRANSFER switch in EMER and then cycle EC Condensate Return IV (39-05).

C. Place EMERGENCY COOLING ISOLATION BYPASS switch in BYPASS and then cycle EC Steam Supply IV (39-09R).

D. Place EMERGENCY COOLING ISOLATION BYPASS switch in BYPASS and then cycle EC Condensate Return IV (39-05).

Answer: A

Associated objective(s):

Development Area (FIO)

Question 7 Details

Question Type:	Multiple Choice
Topic:	NRC RO 7
System ID:	21007
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-207-1-01, EO-1.7

Answer: a. Since the EC auto initiated (>1080 psig for 12 seconds) before control was taken at RSP 11, operation of the EC Condensate Return valve (the desired means) is unavailable therefore the only method to reduce cool down rate is to throttle on the steam supply. To operate the EC from the RSP the CHANNEL 11 CONTROL TRANSFER switch is placed in EMER. The EMERGENCY COOLING ISOLATION BYPASS switch is not operated unless isolation occurs. There is no isolation condition present.

References Provided: None

Question 7 Cross References (table item links)

10CFR55
- 41(b)(7)

Cognitive Level
- 2

Difficulty Level
- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- 295016 AA1.07 4.2/4.3, Rev. NA
AA1.07 Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT.
Control room/local control transfer mechanisms.

Question Source
- New

PROC
- N1-OP-13 Rev. NA
- N1-SOP-9.1 Rev. NA

Question Setting
- C1 (License class closed reference)

8 **SYSID: 21008** **Points: 1.00**

The plant was operating at 100% power when SOP-11.1, RBCLC Failure, is entered. The following conditions now exist:

- N1-SOP-11.1 is still being implemented
- RBCLC system flow and pressure have now been stabilized — sp
- Recirc flow has been reduced to 40 Mlbm/hr
- House service loads have been transferred to Reserve Power

Which one of the following RBCLC heat loads is the most limiting and requires the closest monitoring under these conditions until SOP-11.1 can be exited?

A. Fuel Pool Heat Exchangers.

B. Instrument Air Compressor Inter-and After Coolers.

C. FW Booster Pump Oil Coolers.

D. Off-Gas Vacuum Pump Coolers.

Answer: C

Associated objective(s):

Development Area (FIO)

Question 8 Details

Question Type:	Multiple Choice
Topic:	NRC RO 8
System ID:	21008
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-342-1-01, EO-1.2

N1-SOP-11.1, Table 11.1

Answer: c. SOP 11.1, Table 11.1, designates FWBP oil as major load. All other loads in distractors are all minor.

References Provided: NONE

Question 8 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 2

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 295018 AK1.01 3.5/3.6 Effects on component/system operations
AK1.01 Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Effects on component/system operations.

Question Source

- Bank

PROC

- N1-SOP-11.1, Rev. NA

Question Setting

- C1 (License class closed reference)

9 **SYSID: 21009** **Points: 1.00**

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

- A leak has developed in the 4-inch instrument Air Header
- Instrument Air Pressure is 91 psig and lowering slowly

Assuming no operator action, which one of the following identifies the expected plant response?

A. Backup compressor will auto-start at 90 psig

B. Standby compressor will auto-start at 85 psig

C. Service air cross-connect, BV 94-19 will open at 90 psig

D. Breathing air cross-connect, BV 114-02 will open at 85 psig

Answer: C

Associated objective(s):

Development Area (FIO)

Question 9 Details

Question Type:	Multiple Choice
Topic:	NRC RO 9
System ID:	21009
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-278-1-01, EO-1.4 Reference(s): N1-OP-20 H.2.0 Distractors: No changes - This question tests the ability of the student to recall the correct setpoint for automatic operations of IAS components. All setpoints provided are correct under different conditions than presented in the question. **References Provided: NONE**

Question 9 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 1

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- 295019 AA2.01 3.5/3.6 Instrument air system pressure

AA2.01 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR:
Instrument air system pressure.

Question Source
- Bank

PROC
- N1-OP-20 Rev. NA

Question Setting
- C1 (License class closed reference)

10 **SYSID: 21010** **Points: 1.00**

The plant is in cold shutdown. An inadvertent closure of a shutdown cooling isolation valve results in a loss of shutdown cooling.

Which one of the following is required to prevent thermal stratification?

A. Place one Reactor Recirculation loop in service.

B. Manually initiate one or both Emergency Condensers.

C. Ensure Vessel level is maintained above the Main Steam Line nozzles.

D. Establish RBCLC flow to the Cleanup non-regenerative heat exchanger.

Answer: A

Associated objective(s):

Development Area (FIO)

Question 10 Details

Question Type:	Multiple Choice
Topic:	NRC RO 10
System ID:	21010
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-205-1-01, EO-1.6

Answer: a.N1-OP-4, Section D. Precautions and Limitations, #s 8.0 and 17.0 are the operators responsibilities for preventing thermal stratification in the RPV. The requirements are to have one Reactor Recirculation loop in service **or** shutdown cooling in service **with** RPV water level above the Main Steam Line Nozzles. With no shutdown cooling starting a recirculation loop is required to prevent thermal stratification.

Distractor: b.Manually initiating one or both Emergency Condensers will remove heat if boiling begins in the core but it will not prevent thermal stratification.

Distractor: c.Ensuring Vessel level is maintained above the Main Steam Line nozzles is only successful with Shutdown Cooling in service. Just raising RPV water level above the MSLs will not prevent thermal stratification.

Distractor: d.Establishing flow through the Cleanup system with maximum non-regen flow will remove heat but it will not prevent thermal stratification.

References Provided: None

Question 10 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 1

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- G2.2.12 3/3.4 Knowledge of surveillance procedures
- 295021 Loss of Shutdown Cooling

Question Source
- New

PROC
- N1-OP-4 Rev. NA

Question Setting
- C1 (License class closed reference)

11 **SYSID: 21011** **Points: 1.00**

During refueling operations, an irradiated fuel assembly is dropped onto the bulkhead area of the reactor cavity. No fuel damage occurs.

Which one of the following concerns requires immediate action?

A. Reactor coolant activity levels.

B. Loose parts in the reactor vessel.

C. Airborne radiation on the refuel floor.

D. Radiation levels in the upper areas of the drywell.

Answer: D

Associated objective(s):

Development Area (FIO)

Question 11 Details

Question Type:	Multiple Choice
Topic:	NRC RO 11
System ID:	21011
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-234-1-01, EO-

Answer:	d. Radiation levels in the upper areas of the drywell. Per N1-FHP-25, Precautions and Limitations 4.1.11, highly activated components shall not be placed on the bulkhead areas of the reactor cavity because of the extremely high dose rates this creates in the upper areas of the drywell. In an irradiated fuel assembly was dropped there personnel must be immediately warned of the high dose rates.
Distractor:	a. Reactor coolant activity levels would not be effected because there was no fuel damage, even if activity levels did rise there would not be any immediate concerns.
Distractor:	b. Loose parts in the reactor vessel would not be a concern because there was no fuel damage.
Distractor:	c. Airborne radiation on the refuel floor would not be a concern because there should be no gas emitted from the fuel bundle.

References Provided: None

Question 11 Cross References (table item links)

10CFR55

- 41(b)(8)
- 41(b)(9)
- 41(b)(10)

Cognitive Level

- 2

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 295023 AK1.01 3.6/4.1 Radiation exposure hazards
AK1.01 Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS: Radiation exposure hazards.

Question Source

- New

PROC

- N1-FHP-25 Rev. NA

Question Setting

- C1 (License class closed reference)

12 **SYSID: 21012** **Points: 1.00**

The CRS announces entry into EOP-8, RPV Blowdown because containment parameters are challenging the Pressure Suppression Pressure capability.

Which one of the following is the reason for RPV Blowdown at this time?

Depressurize the reactor while the:

A. torus is still available as a heat sink.

B. drywell is still within its design pressure.

C. torus can still remain within its design temperature limit.

D. drywell can still remain within its design temperature limit.

Answer: A

Associated objective(s):

Development Area (FIO)

Question 12 Details

Question Type:	Multiple Choice
Topic:	NRC RO 12
System ID:	21012
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-344-1-04, EO-1.3

Answer: a. Depressurize the reactor while the torus is still available as a heat sink. Going down the primary containment pressure leg of EOP-4, Primary Containment Control, when parameters are going outside the GOOD area of PSP blowdown is required. Assuming torus level is not a problem (although this would still inop the torus as a heat sink) the problem is that rising pressure in the torus and drywell are reducing the free air space of the torus. While sufficient free air space exists (torus still available as a heat sink) the RPV must be blowndown. That is the reason for RPV Blowdown on rising drywell pressure.

Distractor: b.The drywell is still within its design pressure at this time and will remain within its design pressure for some time after these conditions.

Distractor: c.The torus design temperature limit has no relationship with primary containment pressure and torus level. The Heat Capacity Temperature limit compares torus temperature with reactor pressure to determine the capability of the torus to serve as a heat sink. PSP looks at different parameters.

Distractor: d. The drywell design temperature limit has no relationship with primary containment pressure and torus level. This limit may also require an RPV Blowdown but drywell temperature is not a factor in evaluating PSP.

References Provided: ALL EOPs

Question 12 Cross References (table item links)

10CFR55
- 41(b)(5)

Cognitive Level
- 2

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- 295024 EK3.04 3.7/4.1 Emergency depressurization
EK3.04 Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: Emergency depressurization.

Question Source
- New

PROC
- N1-EOP-4 Rev. NA
- N1-ODP-PRO-0305 Rev. NA

Question Setting
- C1 (License class closed reference)

13 **SYSID: 21013** **Points: 1.00**

Given these conditions:

- The plant is operating at 100% power
- EPR is in control with a setpoint of 920 psig
- MPR setpoint is 930 psig
- Both regulators are sensing 960 psig

A failure of the EPR causes the EPR to sense pressure at 970 psig and rising.

Which one of the following describes main turbine control response?

A. MPR takes control and opens TCVs until 1060 psig.

B. MPR takes control and maintains pressure at 970 psig.

C. EPR remains in control and closes TCVs until 1080 psig.

D. EPR remains in control and lowers pressure to 850 psig.

Answer: D

Associated objective(s):

Development Area (FIO)

Question 13 Details

Question Type:	Multiple Choice
Topic:	NRC RO 13
System ID:	21013
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-248-1-01, EO-1.8

Answer: d. Because the pressure regulator with the highest valve demand (lowest setpoint) is in control the EPR will remain in control and TCVs open in response to the sensed rising pressure this will cause the EPR to open the TCVs (and eventually the BPVs). This will lower pressure until the 850 psig MSIV isolation.

Distractor: a.MPR takes control and opens TCVs until 1060 psig. This response is based on the upper limit of the MPR. The MPR will not take control because it's output is limited to the 30 psi d/p it senses between its setpoint and sensed pressure.

Distractor: b.MPR takes control and maintains pressure at 970 psig. This response is based on the MPR taking control from the EPR. The MPR will not take control from the EPR

Distractor: c. EPR remains in control and closes TCVs until 1080 psig. This is based on the EPR closing TCVS until a reactor scram occurs. The EPR will be opening TCVs on a sensed higher pressure, lowering RPV pressure.

References provided: None

Question 13 Cross References (table item links)

10CFR55
- 41(b)(7)

Cognitive Level
- 2

Difficulty Level
- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- 295025 EK2.08 3.7/3.7 Reactor/turbine pressure regulating system: Plant-Specific
EK2.08 Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: Reactor/turbine pressure regulating system.

Question Source
- New

Question Setting
- C1 (License class closed reference)

14 **SYSID: 21014** **Points: 1.00**

The plant was operating at 100% power, when an ERV opened and remained stuck open. Which of the following identifies the correct EOP action and the bases for this action?

A. Before 95° F in the torus initiate a plant shutdown. This ensures torus temperature does not exceed the upper limit of the Containment Spray NPSH requirements in the event of an accident.

B. Before 100° F in the torus, place Containment Spray in service for torus cooling. To prevent exceeding 110°F in the torus.

C. Before 105° F in the torus initiate a power reduction to prevent the eventual exceeding of the Heat Capacity Temperature Limit.

D. Before 110° F in the torus, place mode switch in shutdown. To reduce the rate of energy production and thus the heat input to the torus.

Answer: D

Associated objective(s):

Development Area (FIO)

Question 14 Details

Question Type:	Multiple Choice
Topic:	NRC RO 14
System ID:	21014
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-344-1-04, EO-1.3

Answer: d. At 110° F in the torus, place mode switch in shutdown. To reduce the rate of energy production and thus the heat input to the torus. This is the EOP required action and is the only torus temperature mentioned in the EOP. The basis for the 110 limit is specified in EOP basis N1-ODP-PRO-0305.

Distractor: a. Before 95° F in the torus initiate a plant shutdown. This ensures torus temperature does not exceed the upper limit of the Containment Spray NPSH requirements in the event of an accident. This is not an EOP action and the bases is not associated with the temperature.

Distractor: b. Before 100° F in the torus, place Containment Spray in service for torus cooling. To prevent exceeding 110°F in the torus. This is not an EOP action and the bases is not associated with the temperature.

Distractor: c. Before 105° F in the torus initiate a power reduction to prevent the eventual exceeding of the Heat Capacity Temperature Limit. This is not an EOP action and the bases is not associated with the temperature.

References Provided: ALL EOPs

Question 14 Cross References (table item links)

10CFR55

- 41(b)(5)

Cognitive Level

- 1

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 295026 EK3.05 3.9/4.1 Reactor SCRAM
EK3.05 Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Reactor SCRAM.

Question Source

- New

PROC

- N1-ODP-PRO-0305 Rev. NA
- N1-SOP-1.4 Rev. NA

Question Setting

- C1 (License class closed reference)

15 **SYSID: 21015** **Points: 1.00**

A LOCA has occurred and the following conditions exist:

- Drywell pressure is 3.0 psig
- Torus pressure is 6.0 psig
- RPV water level is +30 inches
- RPV pressure is 43 psig
- Drywell temperature is 180°F
- Core Spray pumps 112 and 122 are injecting into the RPV
- Containment Spray pumps 111 and 112 are spraying the containment

Which one of the following actions is required?

A. Secure one core spray pump not both.

B. Secure both core spray pumps.

C. Secure one containment spray pump not both.

D. Secure both containment spray pumps.

Answer: D

Associated objective(s):

Development Area (FIO)

Question 15 Details

Question Type:	Multiple Choice
Topic:	NRC RO 15
System ID:	21015
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-344-1-04, EO-1.2

Answer:	d. Secure both containment spray pumps. At less than 3.5 psig in the drywell EOP-4 and EOP-1 require securing containment spray.
Distractor:	a. Because RPV water level is above their initiation setpoint and RPV pressure it is low appears that a core spray pump could be secured, but core spray cannot be secured or throttled until RPV water level is restored above +53 inches.
Distractor:	b. Because RPV water level is above their initiation setpoint it and RPV pressure is low it appears that core spray could be secured, but core spray cannot be secured or throttled until RPV water level is restored above +53 inches.
Distractor:	c. Securing just one containment spray pump will not secure containment spray as required by EOP-4

References Provided: ALL EOPs

Question 15 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 2

Difficulty Level

- Level 4: Highest order knowledge item requiring use of problem solving skills, judgement, and maximum task complexity as well as applying procedures to determine correct answer.

NUREG 1123 KA Catalog Rev. 2

- *G2.4.6 3.1/4 Knowledge symptom based EOP mitigation strategies*

- 295028 High Drywell Temperature

Question Source

- New

PROC

- N1-EOP-1 Rev. NA
- N1-EOP-4 Rev. NA

Question Setting

- C1 (License class closed reference)

16 **SYSID: 21016** **Points: 1.00**

A seismic event has resulted in a reactor scram and torus leak. Torus makeup per EOP-1 Attachment 18 is required. Which one of the following describes how this lineup is accomplished?

A. Align the condensate transfer system to discharge into the core spray keep fill system then into the torus through the core spray test valve.

B. Lineup a containment spray raw water pump discharging into the core spray topping pump discharge then into the torus through the core spray test valve.

C. Align the condensate transfer system to discharge into the containment spray keep fill system then through the containment spray bypass into the torus.

D. Lineup a containment spray raw water pump discharge into the containment spray pump discharge then through the containment spray bypass line into the torus.

Answer: D

Associated objective(s):

Development Area (FIO)

Overlap Scenario #1 Event 07

Question 16 Details

Question Type:	Multiple Choice
Topic:	NRC RO 16
System ID:	21016
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-344-1-23, EO-1.4

Answer: d. Per N1-EOP-1, Attachment 18 line a containment spray raw water pump discharging into the discharge of a containment spray pump then through the containment spray bypass into the torus.

Distractor: a. Although condensate transfer can be used to makeup to the torus (N1-EOP-1, Attachment 6) this lineup does not provide the large amount of flow provided by Attachment 18.

Distractor: b. The containment spray raw water pump does makeup to the torus but the lineup is through the containment spray system, not core spray, although this lineup can be accomplished.

Distractor: c. Although condensate transfer can be used to makeup to the torus (N1-EOP-1, Attachment 6) this lineup does not provide the large amount of flow provided by Attachment 18. Additionally this lineup is through the core spray system not the containment spray system, although this lineup can be accomplished.

References Provided: ALL EOPs

Question 16 Cross References (table item links)

10CFR55

- 41(b)(2)

Cognitive Level

- 1

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- G2.1.23 3.9/4 Ability to perform specific system and integrated plant procedures during different modes of plant operation
- 295030 Low Suppression Pool Wtr Lvl

Question Source

- New

PROC

- N1-EOP-1 Rev. NA

Question Setting

- C1 (License class closed reference)

17 **SYSID: 21017** **Points: 1.00**

A plant transient has resulted in the following conditions:

- Reactor pressure 150 psig
- No ERVs open
- RPV water level -118"
- The only injection source is one (1) Control Rod Drive (CRD) Hydraulic Pump

Which one of the following is the condition of ADEQUATE CORE COOLING (ACC) with these conditions present?

A. There is NO assurance of ACC.

B. Core submergence ensures ACC

C. Steam cooling with injection ensures ACC.

D. Steam cooling WITHOUT injection ensures ACC.

Answer: A

Associated objective(s):

Development Area (FIO)

Question 17 Details

Question Type:	Multiple Choice
Topic:	NRC RO 17
System ID:	21017
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-344-1-01, EO-1.3

Answer: a. is correct. Adequate Core Cooling cannot be assured because none of the mechanisms exist for ACC. RPV level is below Top of Active Fuel (TAF is -84 inches) and RPV level is below Minimum Steam Cooling RPV Water Level (-109 inches) with injection from CRD.

Distractor: b. incorrect. Core submergence does not exist with RPV water level below Top of Active Fuel (TAF is -84 inches)

Distractor: c. incorrect. Steam Cooling with injection is employed in EOP-2, RPV Control only if RPV water level is above Minimum Steam Cooling RPV Water Level (-109 inches), which it is not, with level at -119 inches. (EOP Bases, Definitions page 61)

Distractor: d. incorrect. Steam Cooling without injection is employed in EOP-9, Steam Cooling. With RPV water level below Minimum Steam Cooling RPV Water Level (-109 inches) and any injection source is lined up (one CRD pump) then ACC does not exist by Steam Cooling without injection, since EOP-9 Steam Cooling is not entered under these conditions. If entered override will direct exiting, with any injection source injecting. "ACC cannot be assured if RPV water level is below Minimum Steam Cooling RPV Water Level (-109 inches) and water is being injected into the RPV". (EOP Bases, EOP-9, page 281)

References Provided: All EOPs

Question 17 Cross References (table item links)

10CFR55

- 41(b)(8)

Cognitive Level

- 2

Difficulty Level

- Level 4: Highest order knowledge item requiring use of problem solving skills, judgement, and maximum task complexity as well as applying procedures to determine correct answer.

NUREG 1123 KA Catalog Rev. 2

- 295031 EK1.01 4.6*/4.7* Adequate core cooling
EK1.01 Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: Adequate core cooling.

Question Source

- New

PROC

- N1-EOP-2 Rev. NA
- N1-EOP-9 REV. NA

Question Setting

- C1 (License class closed reference)

18 **SYSID: 21018** **Points: 1.00**

The plant is in an ATWS with the following conditions established:

- RPV level is being maintained between -70 inches and -109 inches
- Liquid Poison (LP) injected using SYS 11
- LP Tank Level recorded at 1400 gallons when LP injection initiated

Which one of the following LP Tank Levels is the MAXIMUM LEVEL that when achieved allows restoration of RPV level to above +53 inches?

A. 1050 gallons.

B. 800 gallons

C. 600 gallons

D. 350 gallons.

Answer: B

Associated objective(s):

Development Area (FIO)

Question 18 Details

Question Type:	Multiple Choice
Topic:	NRC RO 18
System ID:	21018
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-344-1-03, EO-1.2

Answer:	b. Hot shutdown boron weight (600 gallons) must be injected and then RPV level can be raised to above +53 inches. Hot shutdown boron weight (600 gallons) subtracted from the tank volume when LP injection initiated (1400 gallons) equates to 800 gallons (1400 gallons - 600 gallons = 800 gallons)
Distractor:	a. 1050 gallons is the total volume to be injected which equates to cold shutdown boron weight. Only the hot shutdown boron weight must be injected
Distractor:	c. 600 gallons is the total volume to be injected which equates to hot shutdown boron weight. (1400 gallons - 600 gallons = 800 gallons
Distractor:	d. Cold shutdown boron weight (1050 gallons) is the incorrect value. Subtracted from the tank volume when LP injection initiated (1400 gallons) equates to 350 gallons. (1400 gallons – 1050 gallons = 350 gallons)

References Provided: All EOPs

Question 18 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 2

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 295037 EA2.03 4.3*/4.4* SBLC tank level
EA2.03 Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: SBLC tank level.

Question Source

- New

PROC

- N1-EOP-3 Rev. NA

Question Setting

- C1 (License class closed reference)

19 **SYSID: 21019** **Points: 1.00**

A release of radioactivity is in progress. The following radiological conditions are observed.

- Main Stack OGESMS shows rising radiation levels.
- Turbine Building Ventilation PING, shows elevated radiation levels
- Reactor Building Ventilation radiation monitors are reading normal.

Which one of the following describes the probable source of the release?

A. Fuel Clad failure release thru Offgas.

B. Main Steam leakage outside the Primary Containment.

C. Recirculation Pump seal leakage with Primary Containment leakage.

D. Reactor Water Cleanup leakage outside the Primary Containment.

Answer: B

Associated objective(s):

Development Area (FIO)

Question 19 Details

Question Type:	Multiple Choice
Topic:	NRC RO 19
System ID:	21019
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Created from 1998 NRC 18322 ILO **References Provided: NONE**

Question 19 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 1

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 295038 EA2.04 4.1*/4.5* Source of off-site release
EA2.04 Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Source of off-site release.

Question Source

- Bank

Question Setting

- C1 (License class closed reference)

20 **SYSID: 21020** **Points: 1.00**

The plant is at 100% power when the following annunciators are received on the Control Room Main Fire Panel 2-2:

- 2-2-5-2, DSL FIRE PUMP #1 LOW STARTING AIR-FUEL OIL
- 2-2-1-2, DIESEL FIRE PUMP #1 RUNNING
- NO other alarms are received

Which one of the following is the cause of Diesel Fire Pump #1 start?

A. Fuel system is leaking causing a low day tank level.

B. Air system is leaking causing a low starting air pressure.

C. Deluge system initiated beyond the diesel fire pump capacity.

D. Deluge system initiated beyond the electric fire pump capacity

Answer: B

Associated objective(s):

Development Area (FIO)

Question 20 Details

Question Type:	Multiple Choice
Topic:	NRC RO 20
System ID:	21020
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-286-1-01, EO-1.7

Answer:	b. DFP starts on low air start pressure of 73 psig. Alarm on low starting air pressure is 80 psig. Alarm on low day tank level is 175 gallons but does not cause a start of the DFP.
Distractor:	a. Alarm on low day tank level is 175 gallons but does not cause a start of the DFP. Only 2-2-5-2 would be received
Distractor:	c.If a fire system actuated causing a start of the DFP then the electric fire pump would also be running and alarm 2-2-2-2 ELECTRIC FIRE PUMP #1 STARTED would be in alarm. The electric fire pump starts first and then the DFP is system pressure lowers below 100 psig. Suppression system actuation causing fire pump start(s) not cause 2-2-5-2 to alarm.
Distractor:	d.With the electric fire pump running alarm 2-2-2-2 ELECTRIC FIRE PUMP #1 STARTED would be in alarm also. Also suppression system actuation causing fire pump start(s) not cause 2-2-5-2 to alarm.

References Provided: None

Question 20 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 2

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- G2.4.31 3.3/3.4 Knowledge of annunciators alarms and indications, and use of the response instructions 2.4.31 Knowledge of annunciators alarms and indications / and use of the response instructions.

Question Source

- New

PROC

- N1-ARP-A2,2-2 Rev. na
- N1-ARP-A2-2-5-2, Rev. NA
- N1-ARP-A2-2-1-2, Rev. NA

Question Setting

- C1 (License class closed reference)

21 **SYSID: 21021** **Points: 1.00**

A scram from 100% power occurs, with the following: sequence of events

- FWP 11 pump is running in the HPCI mode with its control switch RED flagged.
- FWP 12 pump is running in the HPCI mode with its control switch GREEN flagged.
- RPV water level is 60 inches and rising.
- RO depresses the FEEDWATER RETURN TO NORMAL AFTER HPCI CH 11 & 12 pushbuttons.
- RPV water level rises to +95 inches.
- When RPV water level subsequently lowers to +80 inches, BOTH Motor Feedwater Pump High Level Trip Bypass switches are placed to Bypass and then back to Normal.
- FWP 11 ~~unexpectedly~~ starts.

Which one of the following operator actions would have prevented FWP 11 start, per N1-SOP-1 Reactor Scram?

A. Operating FWP must be placed in PTL prior to resetting HI LEVEL trip.

B. Level Bypass Switches must be left in Bypass until level is in the normal band.

C. HPCI Logic must be reset by placing FWP control switches to Stop spring return to Normal.

D. FWP FCV must be fully closed or the open demand signal will cause a pump start.

Answer: C

Associated objective(s):

Development Area (FIO)

Question 21 Details

Question Type:	Multiple Choice
Topic:	NRC RO 21
System ID:	21021
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-342-1-01, EO-1.2

Answer: c. The HPCI Logic must be reset by placing the FWP control switches Stop spring return to Normal.

Distractor: a. Placing the operating FWP in PTL prior to resetting HI LEVEL trip will prevent the start however, it is not part of the procedural response.

Distractor: b. Leaving Level Bypass Switches in Bypass until level is in the normal band is not part of the procedure – these switches are repositioned only momentarily.

Distractor: d. The FWP FCV must be fully closed or the open demand signal will cause a pump start – the position of the FCV has no input to the pump start logic.

References Provided: None

Question 21 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 1

DER
- NM-2004-3961, Rev. NA

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- G2.1.23 3.9/4 Ability to perform specific system and integrated plant procedures during different modes of plant operation
- 295008 High Reactor Water Level

Question Source
- New

PROC
- N1-SOP-1 Rev. NA

Question Setting
- C1 (License class closed reference)

22 **SYSID: 21022** **Points: 1.00**

The plant is at 100% power when the following alarm is received:

- F2-3-3, REACT VESSEL LEVEL HIGH-LOW
- Level indicating meters on E panel and F panel are reading 74 inches and steady

Per the annunciator response procedure, which one of the following actions is required in response to the alarm?

A. Take manual control of FCV #13 to maintain reactor water level until the cause is corrected.

B. Take manual control of the FW Master Level Controller to maintain reactor water level until the cause is corrected.

C. Have I&C determine the cause of the alarm since it CANNOT be diagnosed using available control room indications.

D. Have I&C fix the RPV level recorder which has an erroneous reading that can be diagnosed based on control room indications.

Answer: D

Associated objective(s):

Development Area (FIO)

Question 22 Details

Question Type:	Multiple Choice
Topic:	NRC RO 22
System ID:	21022
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-259-1-02, EO-1.7

Answer:	d. HIGH/LOW alarm is from the RPV Level Recorder output. With the other instruments indicating normal, the recorder indication is erroneous and has caused the alarm. LR-36-98(ID14) 83" (High) LR-36-98(ID14) 65" (Low)
Distractor:	a. Take manual control of mis-operating systems that are feeding or draining the vessel. FCV#13 is operating correctly.
Distractor:	b. Take manual control of mis-operating systems that are feeding or draining the vessel. Master FW controller is functioning properly
Distractor:	c. With the other instruments indicating normal, the recorder indication is erroneous and has caused the alarm. This can be diagnosed using the control room indications

References Provided: None

Question 22 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 1

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- G2.1.32 3.4/3.8 Ability to explain and apply system limits and precautions
- 295009 Low Reactor Water Level

Question Source
- New

PROC
- N1-ARP-F2-3-3, Rev. NA

Question Setting
- C1 (License class closed reference)

23 **SYSID: 21023** **Points: 1.00**

During conduct of the EOPs, the following conditions exist.

- Reactor pressure is 50 psig
- Drywell pressure is 8 psig
- Drywell temperature elevation 319' is 302°F
- Drywell temperature elevation 263' is 275°F
- Drywell temperature elevation 230' is 250°F
- Drywell bulk average temperature is 270°F

are

In addition to the fuel zone instruments, which one of the following level instruments ~~is~~ available (if any) if actual reactor water level is zero (0) inches?

A. Lo-Lo-Lo instruments.

B. Wide Range instruments.

C. Hi/Lo-Lo/Lo Rosemount instruments.

D. No other instruments can be used.

Answer: D

Associated objective(s):

Development Area (FIO)

Question 23 Details

Question Type:	Multiple Choice
Topic:	NRC RO 23
System ID:	21023
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-344-1-04, EO-1.2

Answer: d. Per EOP Detail A: For all instruments except fuel zone, do not use if drywell temperature near the instrument runs (319' elevation) is at or above the RPV Saturation Temperature (Figure B) OR the instrument reads at or below the minimum usable level.
The 319'elevation temperature at 300°F concurrent with reactor pressure at 50 psig is within the BAD region of the RPV Saturation Temperature (Figure B) therefore other level instruments are unavailable. The other temperatures indicated are within the GOOD region of the RPV Saturation Temperature (Figure B) and therefore could be used if above the minimum usable level such as the Lo-Lo-Lo instruments.

Distractor: a. Lo-Lo-Lo instruments are above the minimum usable level but cannot be used because within the BAD region of the RPV Saturation Temperature (Figure B). Three of the four drywell temperatures are within the GOOD region of the RPV Saturation Temperature (Figure B).

Distractor: b. Wide Range instruments are below the minimum usable level and cannot be used. Also within the BAD region of the RPV Saturation Temperature (Figure B). Three of the four drywell temperatures are within the GOOD region of the RPV Saturation Temperature (Figure B).

Distractor: c. Rosemount instruments are below the minimum usable level but could be inferred as at or above the minimum usable level if the incorrect temperature is used. Also, cannot be used because within the BAD region of the RPV Saturation Temperature (Figure B). Three of the four drywell temperatures are within the GOOD region of the RPV Saturation Temperature (Figure B).

References Provided: EOP-4

Question 23 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 3

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 295012 AA2.01 3.8/3.9 Drywell temperature
AA2.01 Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Drywell Temperature

Question Source

- New

PROC

- N1-ODP-PRO-0305 Rev. NA

Question Setting

- C1 (License class closed reference)

24 **SYSID: 21024** **Points: 1.00**

An ATWS has occurred, with the following:

- Power stabilized at 7%
- Scram air header pressure is 0 psig
- CSO was directed to continue with the actions in EOP 3.1, Alternate Rod Insertion

Which method will be used by the CSO to insert the control rods?

A. Remove the RPS scram fuses

B. Operate Individual Rod Scram Switches

C. Bypass the RWM and drive rods manually

D. Drive control rods by raising cooling water pressure

Answer: C

Associated objective(s):

Development Area (FIO)

Question 24 Details

Question Type:	Multiple Choice
Topic:	NRC RO 24
System ID:	21024
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-344-1-11, EO-1.2

Answer:	c. Bypass the RWM and drive rods manually
Distractor:	a. Remove the RPS scram fuses will have no effect if cause is the hydraulic lock
Distractor:	b. Operating individual Rod Scram switches will have no effect because the scram air header is already depressurized.
Distractor:	d. Drive control rods by raising cooling water pressure may work but will be less effective than using the RMCS due to longer time and raised exposure and

References Provided: None

Question 24 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 1

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 295015 AA1.03 3.6/3.8 RMCS: Plant-Specific
295015 INCOMPLETE SCRAM/1 AA.1.03 Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM: RMCS

Question Source

- New

PROC

- N1-EOP-3.1 Rev. NA

Question Setting

- C1 (License class closed reference)

25 **SYSID: 21025** **Points: 1.00**

A plant startup is in progress, with the following:

- Reactor pressure is at 750 psig
- 08:00: F3-1-2, CONTROL ROD DRIVE PUMP 11 TRIP-VIB
  F3-1-5, CRD CHARGING WTR PRESSURE HI/LO
- CRD Pump 12 will not start
- 08:05: F3-2-5, CRD ACCUMULATOR LEVEL HIGH PRESS LOW, received for one accumulator.

Per N1-SOP-5.1, LOSS OF CRD, which one of the following is the LATEST TIME to insert a manual reactor scram?

A. 08:05

B. 08:10

C. 08:20

D. 08:25

Answer: A

Associated objective(s):

Development Area (FIO)

Question 25 Details

Question Type:	Multiple Choice
Topic:	NRC RO 25
System ID:	21025
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-201-1-01, EO-1.7

Answer:	a. 08:05. With reactor pressure below 900 psig, a scram must be inserted immediately upon receipt of the first accumulator trouble alarm if no CRD pumps are running. Above 900 psig, 20 minutes are permitted from receipt of the first accumulator trouble alarm with no CRD pumps running until the scram is required
Distractor:	b.Made up this time for balance and other SOP actions that are to be taken within 5 minutes
Distractor:	c&d Considers 20 minute time which applies only if above 900 psig and the 20 minute time starts upon receipt of the first accumulator trouble alarm not the CRD pump loss.

References Provided: None

Question 25 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 2

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 295022 AA1.02 3.6/3.6 RPS
AA1.02 Ability to operate and/or monitor the following as they apply to LOSS OF CRD PUMPS: RPS

Question Source

- New

PROC

- N1-SOP-5.1 Rev. NA

Question Setting

- C1 (License class closed reference)

26 **SYSID: 21026** **Points: 1.00**

Correct EOP BASES

An unisolable steam leak has developed in the reactor building with a general area temperature reading 142° F. EOP-5 requires a scram to be inserted. Which one of the following identifies the reason for inserting the scram?

A. Ensure reactor is shutdown prior to leak getting larger.

B. Minimize the effects on Reactor Level Instrumentation.

C. Rapidly lower the leak rate into the secondary containment.

D. Reduce the heat input into the secondary containment.

Answer: D

Associated objective(s):

Development Area (FIO)

Question 26 Details

Question Type:	Multiple Choice
Topic:	NRC RO 26
System ID:	21026
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-344-1-05, EO-1.3

Answer:	d. Reduce rate of heat input into the secondary containment.
Distractor:	a. Ensure reactor is shutdown prior to leak getting larger. This has nothing to do with the bases for the actions
Distractor:	b. Minimize the effects on Reactor Level Instrumentation. Level instruments are not affected by this leak and esp at this temperature
Distractor:	c. Following a scram, the reactor would still be at pressure. The blowdown will reduce leak rate.

References Provided: None

Question 26 Cross References (table item links)

10CFR55
- 41(b)(5)

Cognitive Level
- 1

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- 295032 EK3.02 3.6/3.8 Reactor SCRAM
Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Reactor Scram

Question Source
- New

PROC
- N1-EOP-5 Rev. NA
- N1-ODP-PRO-0305 Rev. NA

Question Setting
- C1 (License class closed reference)

27 **SYSID: 21027** **Points: 1.00**

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

- H2-2-1, R BLDG FL DR SUMPS 11-16 AREA WTR LVL LEVEL HIGH, alarms
- Computer printout confirms the alarm

Which one of the following describes the required operator action?

A. Verify affected sump pump is running and direct Radwaste to determine cause of high level

B. If affected sump rate of rise exceeds 4.2 gpm, manually scram and enter N1-SOP-1

C. Direct Radwaste to pump down affected sump with temporary pump per N1-OP-10

D. Enter N1-EOP-5, Secondary Containment Control, ~~and determine cause of high level~~

Answer: D

Associated objective(s):

Development Area (FIO)

Question 27 Details

Question Type:	Multiple Choice
Topic:	NRC RO 27
System ID:	21027
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-344-1-05, EO-1.2

This question derived for bank SYSID 17917

Answer: d. correct - per ARP-H2-2-1 actions.

Distractor: a, b, c. incorrect - various actions from similar ARPs involving RB EQPT Drains and Drywell sumps.

References Provided: None

Question 27 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 1

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 295036 EA2.02 3.1/3.1 Water level in the affected area
295036 Secondary Containment High Sump/Area Water Level/5
EA2.02 Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL:
Water level in the affected area

Question Source

- Bank

PROC

- N1-ARP-H2-2-1, Rev. NA

Question Setting

- C1 (License class closed reference)

28 **SYSID: 21028** **Points: 1.00**

The Shutdown Cooling System is being placed in service. The inboard isolation valve (38-01) is open however the Outboard Isolation valve (38-02) valve will NOT open.

Which one of the following is preventing valve 38-02 from opening?

A. Reactor water level is 40".

B. Reactor pressure is 125 psig.

C. 38-02 must be opened before 38-01.

D. Reactor coolant temperature is 365° F.

Answer: B

Associated objective(s):

Development Area (FIO)

Question 28 Details

Question Type:	Multiple Choice
Topic:	NRC RO 28
System ID:	21028
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-205-1-01, EO-1.4

Answer: b. Per OP-4 P&L. 38-01, and 38-02 are interlocked so that only one valve can be opened when Reactor Pressure is above 120 psig. Below 120 psig, both valves can be opened.

Distractor: a. Reactor water level is 40". Low level isolation is 5"

Distractor: c. 38-02 must be opened first – doesn't't matter which is open first, both cannot be open if pressure greater than 120 psig

Distractor: d. Reactor coolant temperature is 365° F. AT 350° F the pump will trip however this has NO effect on position of valves.

References Provided: None

Question 28 Cross References (table item links)

10CFR55

- 41(b)(5)

Cognitive Level

- 1

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 205000 K5.02 2.8/2.9 Valve operation
205000 K5.02 Knowledge of the operational implications of the following concepts as they apply to SDC: Valve Operation

Question Source

- New

Question Setting

- C1 (License class closed reference)

29 **SYSID: 21029** **Points: 1.00**

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

- H3-4-6 FEEDWATER CONTROL SYSTEM TROUBLE is received
- W096 FW INST AC POWER TROUBLE alarms
- Subsequently a turbine trip occurs

Which one of the following identifies the feed system response?

A. FCVs 11 and 12 open.

B. FCVs 11 and 12 close.

C. HPCI FCVs remain in pre-transient position.

D. FW low flow valves open to control RPV level.

Answer: C

Associated objective(s):

Development Area (FIO)

Question 29 Details

Question Type:	Multiple Choice
Topic:	NRC RO 29
System ID:	21029
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-259-1-01, EO-1.4

Answer: c. The valves fail as is until power regained and lockout reset.
Distractor: a. The valves fail as is until power regained and lockout reset.
Distractor: b. The valves fail as is until power regained and lockout reset.
Distractor: d. The valves fail as is until power regained and lockout reset.

References Provided: None

Question 29 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 1

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 206000 K2.01 3.2*/3.3* System valves: BWR-2,3,4
Knowledge of electrical power supplies to the following: system valves

Question Source

- New

PROC

- N1-ARP-H3-4-6, Rev. NA

Question Setting

- C1 (License class closed reference)

30 **SYSID: 21030** **Points: 1.00**

Given the following conditions:

- Both emergency condensers have automatically initiated
- Reactor pressure is 1000 psig and dropping

Which one of the following describes how the operator controls the cool down rate per N1-OP-13, "Emergency Cooling System"? Unless directed by the SSS, the operator:

A. Secures one EC system and maintains the cool down rate less than 75°F/hr.

B. Secures one EC system and maintains the cool down rate less than 100°F/hr.

C. Leaves both EC systems in service and maintains the cool down rate less than 75°F/hr.

D. Leaves both EC systems in service and maintains the cool down rate less than 100°F/hr.

Answer: A

Associated objective(s):

Development Area (FIO)

Question 30 Details

Question Type:	Multiple Choice
Topic:	NRC RO 30
System ID:	21030
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-207-1-01, EO-1.7

This is a duplicate of bank question SYSID 12360

Answer: a. Correct. Per N1-OP-13, H.1.5, one EC system is secured and procedural limit for cooldown rate is 75°F/hr.
Distractor: b. Incorrect. Cooldown rate of 100°F/hr would violate procedural limit.
Distractor: c. Incorrect. One EC is to be secured per the OP.
Distractor: d. Incorrect. Procedure direction is to maintain below 75°F/hr.

References Provided: NONE

Question 30 Cross References (table item links)

10CFR55
- 41(b)(5)

Cognitive Level
- 1

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- 207000 K5.09 3.7/4 Cooldown rate: BWR-2,3
K5.09 Knowledge of the operational implications of the following concepts as they apply to ISOLATION (EMERGENCY) CONDENSER: Cooldown rate: BWR-2,3.

Question Source
- Bank

PROC
- N1-OP-13 Rev. NA

Question Setting
- C1 (License class closed reference)

31 **SYSID: 21031** **Points: 1.00**

EC 11 was in operation when a steam flow isolation occurred. With a valid initiation signal present, the operator:

- places EMERG COOLING CHANNEL 11 isolation bypass switch to BYPASS then back to NORM
- places 39-05, EMERG CNDSR RET ISOLATION VALVE 11, control switch to CLOSE

Which one of the following describes the additional action(s) required to establish circulation between EC11 and the reactor vessel?

A. Place 39-05, EMERG CNDSR RET ISOLATION VALVE 11, control switch to open.

B. Open 39-07R, EC STEAM ISOLATION VALVE 112, and 39-09R, EC STEAM ISOLATION VALVE 111.

C. Return EMERG COOLING CHANNEL 11 isolation bypass switch back to BYPASS.

D. Open 05-11, EMERG COND VENT ISOLATION VALVE 112, and 05-01R, EMERG COND VENT ISOLATION VALVE 111

Answer: B

Associated objective(s):

Development Area (FIO)

Question 31 Details

Question Type:	Multiple Choice
Topic:	NRC RO 31
System ID:	21031
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-207-1-01, EO-1.7

Answer: b. During operation of the emergency cooling loops, steam rises from reactor vessel via piping to the condenser tubes where it gives up heat by boiling the condenser shell water at approximately 5 psig. After the steam condenses, it returns by gravity flow to the suction of a reactor recirculating pump and then to the reactor vessel. In the standby condition, the steam isolation valves are normally open so that the tube bundles are continuously at reactor pressure. **The system is placed into operation by opening the normally-closed condensate return isolation valve** which is DC & AC solenoid air-operated.In response to the isolation the following isolation valves close:

- 05-11, EMERG COND VENT ISOLATION VALVE 112
- 05-01R, EMERG COND VENT ISOLATION VALVE 111
- 39-11R, EMERG CONDSR STM SUPPLY DRAIN IV 111
- 39-12R, EMERG CONDSR STM SUPPLY DRAIN IV 112
- 39-07R, EC STM ISOLATION VALVE 112
- 39-09R, EC STM ISOLATION VALVE 111
- 39-05 EMERG CNDSR COND RET ISOLATION VALVE 11

EMERG COOLING CHANNEL 11 isolation bypass keylock switch is taken to BYPASS then back to NORM. 39-05 EMERG CNDSR COND RET

ISOLATION VALVE 11 control switch is taken to CLOSE to reset the isolation signal, which automatically opens this valve and the following valves:

- 05-11, EMERG COND VENT ISOLATION VALVE 112
- 05-01R, EMERG COND VENT ISOLATION VALVE 111
- 39-11R, EMERG CONDSR STM SUPPLY DRAIN IV 111
- 39-12R, EMERG CONDSR STM SUPPLY DRAIN IV 112

Open 39-07R, EC STEAM ISOLATION VALVE 112, and 39-09R, EC STEAM ISOLATION VALVE 111,to return the EC to service.

Distractor: a. Place 39-05, EMERG CNDSR RET ISOLATION VALVE 11, control switch to open is not required. It auto opened when the control switch was taken to closed to reset the isolation signal. Also, steam isolation valves must be manually opened following an isolation signal to return the EC to service. In standby, the steam isolation valves are open and the action to place the EC in service is to open 39-05 EMERG CNDSR COND RET ISOLATION VALVE.

Distractor: c. Return EMERG COOLING CHANNEL 11 isolation bypass switch back to BYPASS is not required. The operator action is correct. Also, steam isolation valves must be manually opened following an isolation signal to return the EC to service. In standby, the steam isolation valves are open and the action to place the EC in service is to open 39-05 EMERG CNDSR COND RET ISOLATION VALVE.

Distractor: d. Open 05-11, EMERG COND VENT ISOLATION VALVE 112, and 05-01R, EMERG COND VENT ISOLATION VALVE 111 is not required since these valves auto opened, however, the EC is not in service until the steam isolation valves are open. In standby, the steam isolation valves are open and the action to place the EC in service is to open 39-05 EMERG CNDSR COND RET ISOLATION VALVE.

References Provided: None

Question 31 Cross References (table item links)

10CFR55

- 41(b)(5)

Cognitive Level

- 2

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- *207000* A1.03 3.3*/5-*Mar Steam flow:* BWR-2,3
A1.03 Ability to predict and/or monitor changes in parameters associated with operating the ISOLATION (EMERGENCY) CONDENSER controls including:
Steam flow: BWR-2,3.

Question Source

- New

PROC

- N1-OP-13 Rev. NA

Question Setting

- C1 (License class closed reference)

32 **SYSID: 21032** **Points: 1.00**

With the plant at 100% power, operability of the Core Spray System power-operated valves and pumps is required to be demonstrated quarterly

each quarter

Because of the design of the Core Spray System, which one of the following is the number of QUARTERLY SURVEILLANCE TESTS that are performed to demonstrate operability of all required Core Spray System power-operated valves and pumps?

A. 1

B. 2

C. 4

D. 8

Answer: C

Associated objective(s):

Development Area (FIO)

Question 32 Details

Question Type:	Multiple Choice
Topic:	NRC RO 32
System ID:	21032
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-209-1-01, EO-1.4

Answer: c. There are 4 separate quarterly STs (N1-ST-Q1A, Q1B, Q1C, Q1D) which are used to demonstrate operability of the Core Spray System power-operated valves and pumps – a separate test for each Core Spray Subsystem (111, 112, 121, 122) and its associated power-operated valves

Distractor: a. Many of the quarterly operability tests for other plant systems are contained within a single surveillance test making this a plausible distracter. However, CS uses 4 separate tests.

Distractor: b. Core Spray System has system 11 and system 12, therefore 2 is a plausible distracter considering that each system could be tested within a single surveillance test. However, CS uses 4 separate tests

Distractor: d. With four separate subsystems, separate tests could be developed for pumps and for valves for a total of eight surveillance tests which is plausible because of the complex design of the core spray system. However, Core Spray testing combines the pump and power-operated valve testing within one surveillance test.

References Provided: NONE

Question 32 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 1

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 209001 K4.10 2.8/2.9 Testability of all operable components
K4.10 Knowledge of LOW PRESSURE CORE SPRAY SYSTEM design feature(s) which provide for the following: Testability of all operable components.

Question Source

- New

PROC

- N1-ST-Q1A Rev. NA
- N1-ST-Q1B Rev. na
- N1-ST-Q1C, Rev. NA
- N1-ST-Q1D, Rev. NA

Question Setting

- C1 (License class closed reference)

33 **SYSID: 21033** **Points: 1.00**

The plant is at 100% power with Core Spray (CS) surveillance testing in progress. Current Core Spray valves alignment is as follows:

- 40-10, CORE SPRAY DISCHARGE IV 112 (inside) is closed
- 40-11, CORE SPRAY DISCHARGE IV 111 (inside) is closed
- 40-12, CORE SPRAY DISCHARGE IV 11 (outside) is closed
- 40-06, CORE SPRAY TEST VALVE 11 is open

Subsequently, a coolant leak occurs. Current event parameters are:

- Reactor water level is 20 inches
- Reactor pressure is 500 psig
- Drywell pressure is 8 psig

Considering the Core Spray valves identified above, which ~~one of the following describes ALL~~ valve(s) ~~that~~ will receive a signal to reposition based on the current event parameters?

A. 40-06

B. 40-12

C. 40-10, 40-11, 40-12

D. 40-06, 40-10, 40-11

Answer: A

Associated objective(s):

Development Area (FIO)

Question 33 Details

Question Type:	Multiple Choice
Topic:	NRC RO 33
System ID:	21033
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-209-1-01, EO-1.4

Answer: a. Correct. On initiation of Core Spray with Drywell Pressure above 3.5 psig, test return valve 40-06 will close.

Distractor: b, c, &d. Incorrect. With RPV pressure at 500 psig, to Discharge IVs 40-10, 40-11 and 40-12 will not reposition (open) until RPV pressure drops below 365 psig.

References Provided: None

References C19859-c, Sheet 5 G-2, Sheet 9 A-2

Question 33 Cross References (table item links)

10CFR55
- 41(b)(5)

Cognitive Level
- 1

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

DRW
- C-19859-C,(RPS) Rev.

NUREG 1123 KA Catalog Rev. 2
- 209001 A1.08 3.3/3.2 System lineup
A1.08 Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURE CORE SPRAY SYSTEM controls including: System lineup

Question Source
- Modified

Question Setting
- C1 (License class closed reference)

34 **SYSID: 21034** **Points: 1.00**

The following conditions exist:

- The plant has experienced an ATWS from 100% power
- All attempts to scram the reactor have failed
- The CSO is driving rods using CRD Pump 12
- Liquid Poison System 12 has been initiated

The following events then occur:

- **ALL** offsite power is lost
- EDG 103 fails to start
- No actions have been taken respecting the loss of offsite power

Given these conditions, which one of the following identifies the quickest method to restore boron injection?

A. Manually start Liquid Poison Pump 11.

B. Restart Liquid Poison Pump 12.

C. Use reactor water cleanup to inject liquid poison.

D. Use the hydro pump to inject liquid poison.

Answer: A

Associated objective(s):

Development Area (FIO)

Question 34 Details

Question Type:	Multiple Choice
Topic:	NRC RO 34
System ID:	21034
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-211-1-01, EO-1.8

This question was developed using 17194 from Exam Bank

Answer: a. Manually inject with system 11 is **correct** - under these conditions, power is available to PB102 which provides power to LP#11. EDGs automatically power ECCS boards (102 & 103) on a LOOP therefore no operator action beyond placing the control switch in "SYS 11" is required to restore injection.

Distractor: b. EDG 103 failed to start causing a loss of PB103 and a loss of AC power to LP Pump 12.

Distractor: c & d. Alternate Boron Injection is **wrong** - Either method, using the hydro pump or RWCU, requires more time and resource than required to manually initiate SYS 1~~2~~.

References Provided: NONE

11

Question 34 Cross References (table item links)

10CFR55

- 41(b)(5)

Cognitive Level

- 2

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 211000 A2.03 3.2/3.4* A.C. power failures

Question Source

- Modified

Question Setting

- C1 (License class closed reference)

[illegible]

ability to predict the impacts of AC power failures & use procedures to correct control mitigate

all on Modified questions please provide original question

35 **SYSID: 21035** **Points: 1.00**

Following a failure to scram, the Liquid Poison (LP) System is started by placing the key lock selector switch to SYS 11. The following are observed for the LP System at the K Panel:

- System 11 explosive valve continuity light is OFF.
- System 12 explosive valve continuity light is OFF.
- Reactor Water Cleanup is isolated.
- Pump 11 started.
- Pump 11 discharge pressure is oscillating between 950 and 1000 psig.
- Tank level is NOT lowering.

Which one of the following actions is required and why?

A. Start System 12 because System 11 is cycling back to the LP tank.

B. Align the hydro pump for injection because the squibb valves did not fire.

C. Continue injecting with System 11 because there is adequate flow to the Reactor.

D. Align the Cleanup System for injection because liquid poison injection is blocked.

Answer: A

Associated objective(s):

Development Area (FIO)

Question 35 Details

Question Type:	Multiple Choice
Topic:	NRC RO 35
System ID:	21035
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-211-1-01, EO-1.8

Answer: a. Start System 12 because System 11 is cycling back to LP tank. System 11 discharge relief valve is cycling allowing the LP pump discharge to return to the LP tank. Starting the N1-OP-12, requires starting the standby pump. LP pump 12 will bypass the #11 pump relief valve and permit the #12 system to supply boron.

Distractor: b. Aligning the hydro pump is available, but this requires time, additionally N1-OP-12 requires starting the standby pump.

Distractor: c. Adequate flow is not demonstrated because the LP tank level is not lowering.

Distractor: d. Aligning the cleanup system is an option, but this requires time, additionally N1-OP-12 requires starting the standby pump.

References Provided: None

Question 35 Cross References (table item links)

10CFR55
- 41(b)(7)

Cognitive Level
- 2

Difficulty Level
- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- 211000 A4.08 4.2/4.2, Rev. NA
A4.08 Ability to manually operate and/or monitor in the control room: System initiation.

Question Source
- New

12

PROC
- N1-OP-10 Rev. NA

Question Setting
- C1 (License class closed reference)

36 **SYSID: 21036** **Points: 1.00**

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

- Group 1 Lights for RPS Channel 12 on Panels F and M are NOT LIT
- Group 1 light bulbs have been verified working
- All other group lights are LIT
- then APRM Channel 16 fails upscale

Which one of the following identifies the required operator actions?

A. Bypass APRM 16 and reset RPS Channel 12.

B. Bypass APRM 16 and reset RPS Channel 11.

C. Insert a manual scram and implement SOP-1

D. Confirm automatic scram and implement SOP-1.

Answer: A

Associated objective(s):

Development Area (FIO)

Question 36 Details

Question Type:	Multiple Choice
Topic:	NRC RO 36
System ID:	21036
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-212-1-01, EO-1.4

Question derived from bank SYSID 12388

Answer:	a. Correct. APRM 16 provides trip input into RPS Channel 12. With Group 1 RPS Channel 12 lights out, and APRM 16 above its trip setpoint, the result will be only a trip of RPS Channel 12.
Distractor:	b. Incorrect. APRM 16 provides input into RPS Channel 12. RPS Channel 11 will not trip and does not require reset.
Distractor:	c. Incorrect. Manual scram is not required. If the Group 1 lights were NOT LIT on Channel 11 and APRM 16 tripped Channel 12, then some rod motion will occur and a manual scram is required.
Distractor:	d. Incorrect. If Group 1 lights were NOT LIT on Channel 11 and Channel 12 trip, then rod motion may result in an automatic scram.

References provided: NONE

Question 36 Cross References (table item links)

10CFR55
- 41(b)(7)

Cognitive Level
- 2

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- 212000 K1.01 3.7/3.9 Nuclear instrumentation
K1.01 Knowledge of the physical connections and/or cause-effect relationships between REACTOR PROTECTION SYSTEM and the following: Nuclear instrumentation.

Question Source
- Modified

Question Setting
- C1 (License class closed reference)

37 **SYSID: 21037** **Points: 1.00**

A plant startup is in progress, with the following:

- Reactor is critical with a heatup in progress
- IRM 16 detector is inadvertently selected for withdrawal.
- The FULL OUT pushbutton is depressed

Which one of the following describes the IRM 16 response and the effect on control rod withdrawal?

A. Withdraws and a rod block occurs.

B. **NOT** withdraw and a rod block occurs.

C. Withdraws and control rod movement may continue.

D. **NOT** withdraw but control rod movement may continue.

Answer: A

Associated objective(s):

Development Area (FIO)

Question 37 Details

Question Type:	Multiple Choice
Topic:	NRC RO 37
System ID:	21037
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-215-1-02, EO-1.4

NOT

Answer: a. IRM 16 will withdrawal but a control rod block will be generated because the detectors are inserted with the Mode Switch in Startup.

Distractor: b. There is no interlocked to prevent IRM withdrawal.

Distractor: c. A control rod block will be generated because the detector is inserted with the Mode Switch in Startup.

Distractor: d. There is no interlocked to prevent IRM withdrawal and control rod block will be generated because the detector is inserted with the Mode Switch in Startup.

References Provided: None

Question 37 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 2

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 215003 K4.05 2.9/3 Changing detector position
K4.05 Knowledge of INTERMEDIATE RANGE MONITOR (IRM) SYSTEM design feature(s) and/or interlocks which provide for the following: Changing detector position.

Question Source

- New

PROC

- N1-OP-38B Rev. NA

Question Setting

- C1 (License class closed reference)

38 **SYSID: 21038** **Points: 1.00**

A reactor startup is in progress. The reactor is critical and a heatup is in progress. The following conditions exist:

- All IRMs are on Range 5
- SRM 11 now reads 70 cps
- SRM 12 now reads 175 cps
- SRM 13 now reads 250 cps
- SRM 14 now reads 140 cps
- Electrical power is lost to SRM 13 detector drive
- A control rod block exists

Which one of the following is required to clear the existing control rod block?

A. Insert SRM 11.

B. Withdraw SRM 11.

C. Place SRM 13 in Bypass.

D. Restore power to SRM 13.

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

NRC 2004 RO WRITTEN EXAMINATION

Question 38 Details

Question Type:	Multiple Choice
Topic:	NRC RO 38
System ID:	21038
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-215-1-01, EO-1.4

Answer: a. Insert SRM 11 until greater the 100 cps or it's fully inserted. With the SRM fully inserted the downscale setpoint changes from 100 cps to 3 cps.

Distractor: b. Withdrawing SRM 11 will result in an even lower SRM count rate. The rod block will not clear.

Distractor: c. The loss of power to the detector drive does not cause a control rod block. Bypassing SRM 13 will not clear the rod block.

Distractor: d. The loss of power to the detector drive does not cause a control rod block. Restoring power to SRM 13 will not clear the rod block.

References Provided: None

Question 38 Cross References (table item links)

10CFR55

- 41(b)(5)

Cognitive Level

- 2

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 215004 A1.05 3.6/3.8 SCRAM, rod block, and period alarm trip setpoints
  A1.05 Ability to predict and/or monitor changes in parameters associated with operating the SOURCE RANGE MONITOR (SRM) SYSTEM controls including:
  SCRAM, rod block, period alarm trip setpoints.

Question Source

- New

PROC

- N1-OP-38A Rev. NA

Question Setting

- C1 (License class closed reference)

39 **SYSID: 21039** **Points: 1.00**

The plant is at 80% power in five (5) recirculation loop operation. The output of APRM Flow Unit 11 lowers to and remains at zero (0).

Which one of the following is the effect of this malfunction?

A. Rod block. No ½ scram.

B. Rod block and ½ scram.

C. Just a flow unit downscale alarm.

D. Just a flow unit inoperable alarm.

Answer: B

Associated objective(s):

Development Area (FIO)

Question 39 Details

Question Type:	Multiple Choice
Topic:	NRC RO 39
System ID:	21039
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-215-1-02, EO-1.4

Answer:	b. Correct. Flow Unit 11 output is used by RPS Channel 11 APRMS 11, 12, 13, 14. With flow signal of zero into these APRMs and power at 80% an APRM upscale trip occurs, resulting in a RPS Channel 11 trip and APRM upscale rodblock. Power is above the flow biased APRM upscale trip setpoint.
Distractor:	a. Incorrect. RPS Channel 11 trips
Distractor:	c. Incorrect. Additional actions do occur as a result of this downscale failure, such as the rod block and half scram from APRMs.
Distractor:	d. Incorrect. Low flow is not an inop trip. If failure was caused by inop flow unit, the failure will still rsult in a half scram.

References Provided: None

C-19859-C, Sheet 5 G-2, Sheet 9 A-2

Question 39 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 3

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

DRW

- C-19859-C,(RPS) Rev.

NUREG 1123 KA Catalog Rev. 2

- 215005 A3.05 3.3/3.3 Flow converter/comparator alarms
A3.05 Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including: Flow converter/comparator alarms.

Question Source

- New

Question Setting

- C1 (License class closed reference)

EXAMINATION ANSWER KEY

NRC 2004 RO WRITTEN EXAMINATION

40 **SYSID: 21040** **Points: 1.00**

The plant is at 30% power, with the following:

- Four APRM 12 inputs, "are bypassed" (I.E., LPRMS 04-33C, 12-41A, 20-33A and 20-49A) ~~are bypassed~~
- Both LPRM Downscale Buttons have been depressed on "E" Console ?
- LPRM 04-33A input to APRM 12 fails downscale, while buttons are still depressed

Which one of the following describes the condition of APRM 12?

APRM 12 is.........

A. operable with a downscale trip generated.

B. operable with an APRM inop trip generated.

C. inoperable with a downscale trip generated.

D. inoperable with an APRM inop trip generated.

Answer: D

Associated objective(s):

Development Area (FIO)

Question 40 Details

Question Type:	Multiple Choice
Topic:	NRC RO 40
System ID:	21040
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-215-1-02, EO-1.4

Question created from Bank Question SYSID 12438

Answer: d. Correct. APRM 12 will generate an inop trip when LPRM 04-33A fails downscale because the count of available LPRMs has dropped below 4. The APRM must also be declared inoperable.

Distractor: a&c. Incorrect. LPRM downscale will not generate an APRM downscale because the LPRM Downscale buttons are depressed. IF these were not depressed, a downscale trip would be generated.

Distractor: b. Incorrect. APRM will generate an inop trip, but the instrument is inoperable.

References Provided: NONE

Question 40 Cross References (table item links)

10CFR55
- 41(b)(6)

Cognitive Level
- 2

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- G2.1.33 3.4/4 Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications

Question Source
- Modified

PROC
- N1-OP-39 Rev. NA

Question Setting
- C1 (License class closed reference)

EXAMINATION ANSWER KEY

NRC 2004 RO WRITTEN EXAMINATION

41 **SYSID: 21041** **Points: 1.00**

The following conditions ~~exist.~~ ~~have existed for two minutes~~

- A LOCA has occurred
- annunciator MAIN STM LINE AUTOMATIC DE-PRESS TIMING alarmed
- annunciator MAIN STM LINE ELECTROMATIC RELIEF VALVE OPEN alarmed
- Reactor water level is -15 inches and steady
- Reactor pressure is 200 psig
- The Channel 11&12 ADS Reset pushbuttons are depressed

Which one of the following describes the expected response of the ADS ERVs following ADS reset pushbutton being depressed?

A. Close and ~~remain closed~~. No further signal for valves to reopen

B. Close and reopen in 111 seconds.

C. Remain open until reactor pressure < 50 psig.

D. Close and reopen when the RESET pushbuttons are released.

Answer: B

Associated objective(s):

Development Area (FIO)

Confirm -15" is not Low Low Low

Question 41 Details

Question Type:	Multiple Choice
Topic:	NRC RO 41
System ID:	21041
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-218-1-01, EO-1.7

Answer:	b. Resetting the ADS timers restarts the 111 second timers with pressure remaining in the RPV the valves would reopen in 111 seconds.
Distractor:	a. The ERVs will reopen because there is still RPV pressure and lo lo lo level.
Distractor:	c. The ERVs will close when the timer is reset because it interrupts the initiation circuit.
Distractor:	d The ERVs will close for 111 seconds when the timer is reset because it interrupts the initiation circuit and the timer must time out again.

References Provided: None

Question 41 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 2

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- G2.4.31 3.3/3.4 Knowledge of annunciators alarms and indications, and use of the response instructions
2.4.31 Knowledge of annunciators alarms and indications / and use of the response instructions.
- 218000 Automatic Depressurization System

Question Source
- New

PROC
- N1-ARP-F2 Rev. NA

Question Setting
- C1 (License class closed reference)

42 **SYSID: 21042** **Points: 1.00**

The plant is inerting the Drywell per N1-OP-9, Section E.4. The following valves have been opened:

- 201-40, BV-DRYWELL TO RB VENT SYS
- 201-31, DW N2 VENT & PURGE ISOLATION VALVE 12
- 201-32, DW N2 VENT & PURGE ISOLATION VALVE 11
- 201-10, DW AIR VENT & PURGE ISOLATION VALVE 11
- 201.2-624, BV-N2 TO DW & TORUS FROM STM VAPORIZER
- 201-09, DW AIR VENT & PURGE ISOLATION VALVE 12

Which one of the following will occur if plant conditions cause annunciator H1-1-8, STACK GAS MONITORING HIGH RADIATION to alarm?

A. Only 201.2-624 and 201-40 close

B. Only 201-09, 10, 31 and 32 close

C. Only 201-09, 10, 31, 32 and 40 close

D. Only 201.2-624, 09, 10, 31 and 32 close

Answer: B

Associated objective(s):

Development Area (FIO)

Question 42 Details

Question Type:	Multiple Choice
Topic:	NRC RO 42
System ID:	21042
User ID:	
Status:	Active
Must Appear;	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-223-1-04, EO-1.4

Answer: b. Only 201-09, 10, 31 and 32 close. 201.2-624 which supplies N2 from the vaporizer does not close. 201-40 is a manual block valve and does not close.

Distractor: a. 201.2-624 which supplies N2 from the vaporizer does not close. 201-40 is a manual block valve and does not close.

Distractor: c. 201-40 is a manual block valve and does not close.

Distractor: d 201.2-624 which supplies N2 from the vaporizer does not close.

References Provided: None

Question 42 Cross References (table item links)

10CFR55

- 41(b)(5)

Cognitive Level

- 1

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 223002 A1.02 3.7/3.7 Valve closures
A1.02 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.

Question Source

- New

PROC

- N1-ARP-H1 Rev. NA
- N1-OP-09 Rev. NA

Question Setting

- C1 (License class closed reference)

EXAMINATION ANSWER KEY

NRC 2004 RO WRITTEN EXAMINATION

Blue light ind power available Still operable

43 **SYSID: 21043** **Points: 1.00**

The plant is operating at 95% power with the following conditions:

- Both GREEN and RED lights for ERV111 position indication are extinguished
- On "M" Panel, the white test lights for ERV121 are extinguished
- All affected lamps and fuses have been verified good

All other plant indications are normal, no alarms are present

Given these conditions, which one of the following describes the **automatic** capabilities of ERV111 and ERV121?

A. Both ERV111 and ERV121 will function as designed for ADS and pressure relief.

B. Neither ERV111 or ERV121 will function as designed for ADS or pressure relief.

C. ERV111 will function as designed for pressure relief and ADS;
ERV121 will only function as a pressure relief.

D. ERV121 will function as designed for ADS and pressure relief;
ERV111 will only function if manually operated.

Answer: C

Associated objective(s):

Development Area (FIO)

Question 43 Details

Question Type:	Multiple Choice
Topic:	NRC RO 43
System ID:	21043
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-239-1-01, EO-1.4

Question derived from bank 17210

References Provided: None

Question 43 Cross References (table item links)

10CFR55
- 41(b)(7)

Cognitive Level
- 2

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- 239002 K1.08 4.0*/4.1 Automatic depressurization system
K1.08 Knowledge of the physical connections and/or cause-effect relationships between RELIEF/SAFETY VALVES and the following: Automatic depressurization system.

Question Source
- Bank

Question Setting
- C1 (License class closed reference)

44 **SYSID: 21044** **Points: 1.00**

The plant is at 100% power:

- The BLUE LIGHT above the control switches for ERV 111, 112, and 113 is OFF
- The BLUE LIGHT above the control switches for ERV 121, 122, and 123 is ON

Assuming ERVs open at their design set point (±0 psig), which one of the following is the LOWEST reactor pressure at which an ERV opens during an overpressure event?

A. 1090 psig.

B. 1095 psig.

C. 1100 psig.

D. 1105 psig.

Answer: A

Associated objective(s):

Development Area (FIO)

Question 44 Details

Question Type:	Multiple Choice
Topic:	NRC RO 44
System ID:	21044
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-239-1-01, EO-1.4

Answer: a. ERV Indication on F Panel
ERV Indication is located on F Panel and contains the following indication:
• Red and Green lights indicate the position of the pilot valve solenoid.
• The normally lit blue light above each ERV control switch indicates that DC Control Power is available to the ERV pilot valve solenoid. The Blue Light will de-energize for two reasons: (1) If the ERV receives an initiating signal or (2) If DC Control Power to the ERV pilot valve solenoid circuit is lost. Erv 111, 112, and 113 have no DC control power and will NOT function.
• RED status monitoring lights to the right of each ERV Control Switch provides additional indication that an ERV has opened. A high alarm condition, as monitored by the acoustic monitors, will cause the RED light to illuminate. The light will remain illuminated until the alarm is reset at the System Panel.
ERV 111 and 122 open at 1090 psig
ERV 112 and 121 open at 1095 psig
ERV 113 and 123 open at 1100 psig
1105 psig is an arbitrary value based on balance with other answers and also would be indicative that NO ERVs open and safety valves actuate.

Distractors: b.c.d. See justification above.

References Provided: NONE

Question 44 Cross References (table item links)

10CFR55
- 41(b)(7)

Cognitive Level
- 1

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- 239002 K3.02 4.2*/4.4 Reactor over pressurization
K3.02 Knowledge of the effect that a loss or malfunction of the RELIEF/SAFETY VALVES will have on following:
Reactor Over-pressurization

Question Source
- New

Question Setting
- C1 (License class closed reference)

45 **SYSID: 21045** **Points: 1.00**

The plant is at 100% power. The Feedwater Level Control System is maintaining RPV water level at the desired reactor water level in three-element control.

The total main steam flow output to the steam flow-feed flow comparator goes to ZERO and remains at zero.

Which one of the following describes the direction and magnitude of the RPV water level change?

A. **Lowers** and the reactor scrams on low level.

B. **Lowers** but power operation continues. No reactor scram occurs.

C. **Rises** and the turbine trips resulting in a reactor scram.

D. **Rises** but power operation continues. No reactor scram occurs.

Answer: A

Associated objective(s):

Development Area (FIO)

Question 45 Details

Question Type:	Multiple Choice
Topic:	NRC RO 45
System ID:	21045
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-259-1-02, EO-1.8

Answer:	a. With a lower steam flow, the FWLC system senses less a lower demand in FW flow to maintain RPV water level and level lowers. Because of the failure of the total steam flow output signal to zero, the magnitude of the RPV level change demanded is approximately 40 inches. Level will lower below the low level scram setpoint
Distractor:	b.The magnitude of the signal error will cause a reduction in FW flow such that level lowers below the low level scram setpoint.
Distractor:	c. If the steam flow signal output signal increases, then the resulting RPV level will increase. However, with the steam flow output failed high, the magnitude of the level change would not cause a turbine trip.
Distractor:	d. If the steam flow signal output signal increases, then the resulting RPV level will increase

References Provided: None

Question 45 Cross References (table item links)

10CFR55
- 41(b)(7)

Cognitive Level
- 1

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- 259002 K6.03 3.1/3.1 Main steam flow input
K6.03 Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER LEVEL CONTROL SYSTEM: Main steam flow input.

Question Source
- New

Question Setting
- C1 (License class closed reference)

46 **SYSID: 21046** **Points: 1.00**

Reactor power is being adjusted when the following conditions are observed:

- F2-3-3, REACT VESSEL LEVEL HIGH-LOW alarms
- RPV level is observed at 65 inches when the alarm occurs
- RPV level is lowering 0.20 inches per minute
- FCV 13 is stuck at its current position and will NOT respond to control room controls

Each of the following actions is appropriate under these conditions **except** one. Which one of the following actions is the **exception**?

A. Lower reactor power to stabilize level.

B. Observe steam flow higher than feed flow.

C. Transfer 13 FWP FCV to local manual control at TB 305'.

D. Determine why the low level alarm occurred later than expected.

Answer: D

Associated objective(s):

Development Area (FIO)

Question 46 Details

Question Type:	Multiple Choice
Topic:	NRC RO 46
System ID:	21046
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-259-1-02, EO-1.7

Answer: d. Low level alarm actuates at 65" and alarmed when expected. The candidate must determine the direction that reactor power was being adjusted to determine the appropriateness of answers a and b. To receive a low level alarm reactor power was being raised and it is appropriate to lower reactor power to stabilize RPV water level and return it to the normal operating range. With a lowering reactor water level resulting from the power increase and a failure of FCV13 the steam flow would be higher than the feed flow (unchanged) indicating trouble with the feed water level control system.

Distractor: a. b. c. See justification above.

References Provided: NONE

Question 46 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 2

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- G2.4.50 3.3/3 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual
- 259002 Reactor Water Level Control System

Question Source

- New

PROC

- N1-ARP-F2 Rev. NA
- N1-OP-16 Rev. NA

Question Setting

- C1 (License class closed reference)

47 **SYSID: 21047** **Points: 1.00**

N1-OP-9, H.1.0, Venting Primary Containment Through RBEVS During Normal Ops, is in progress venting the torus using RBEVS #11. RBEVS Fan #11 trips.

Which one of the following describes the effect of the fan trip and the action in response to this effect?

A. RBEVS train isolates but primary containment and secondary containment remain in communication until valves are manually closed.

B. RBEVS train isolates and primary containment to secondary containment communication is interrupted by automatic valve closures.

C. RBEVS train is unisolated and primary containment and secondary containment remain in communication until valves are manually closed.

D. RBEVS train is unisolated until valves are manually closed but primary containment to secondary containment communication is interrupted by automatic valve closures.

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

NRC 2004 RO WRITTEN EXAMINATION

Question 47 Details

Question Type:	Multiple Choice
Topic:	NRC RO 47
System ID:	21047
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-261-1-01, EO-1.8

Answer: c. correct. The only interlock is between the inlet and cooling dampers. Fan trip will not close any dampers.

Distractors: a, b, d. incorrect. The purge valves will not close automatically because no containment isolation or hi hi stack radiation signal is present.

References Provided: NONE

Question 47 Cross References (table item links)

10CFR55

- 41(b)(5)

Cognitive Level

- 1

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 261000 A2.05 3/3.1 Fan trips

A2.05 Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Fan trips.

Question Source

- *New*

Question Setting

- C1 (License class closed reference)

48 **SYSID: 21048** **Points: 1.00**

The plant is at 100% power with Motor-Operated Disconnect (MOD) 8106 between South Oswego No. 1 and NMP-Fitzpatrick No. 4 open. Because of a transient the operators insert a manual reactor scram.

Which one of the following describes the AC Distribution System response if the R40 breaker trips and WILL NOT close?

A. EDG 102 and EDG 103 are off. R10 is closed to power PB 11 and PB 12. PB 101 is de-energized.

B. EDG 102 and EDG 103 power PB 102 and PB 103. All other AC busses are de-energized.

C. EDG 103 powers PB 103. Fast transfer powers PB 11 from Transformer T101N.

D. EDG 102 powers PB 102. Fast transfer powers PB 12 from Transformer T101S.

Answer: C

Associated objective(s):

Development Area (FIO)

Question 48 Details

Question Type:	Multiple Choice
Topic:	NRC RO 48
System ID:	21048
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-264-1-01, EO-1.4

Answer:

Distractor: a. R10 will power PB 11 but not PB12. EDG 103 starts to power PB 103.

Distractor: b. PB 11 is powered by R10. Only EDG 103 starts to power PB 103.

Distractor: d. EDG 102 does not start. EDG 103 starts to power PB 103. PB 12 is deenergized.

References Provided: NONE

Question 48 Cross References (table item links)

AC Dist - knowledge of physical connections & cause & effect relationships between AC Dist & EDG's

10CFR55

- 41(b)(7)

Cognitive Level

- 2

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 262001 K1.01 3.8/4.3* Emergency generators (diesel/jet)

Question Source

- Bank

Question Setting

- C1 (License class closed reference)

49 **SYSID: 21049** **Points: 1.00**

A fault occurs within UPS 172 causing the inverter output voltage to lower to one-half of its operating value.

Which one of the following actions is required to assure that the loads supplied by UPS 172 are powered?

A. Confirm loads are powered from I&C Bus 130A with no operator action taken.

B. Manually transfer UPS 172 loads to BYPASS TO LOAD per the operating procedure.

C. Confirm loads are aligned to the BYPASS transformer with no operator action taken.

D. Close the MAINTENANCE breaker then open the NORMAL supply breaker for I&C Bus 130A.

Answer: C

Associated objective(s):

Development Area (FIO)

Question 49 Details

Question Type:	Multiple Choice
Topic:	NRC RO 49
System ID:	21049
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OP-001-262-1-03, EO-1.8

Answer: c. Each UPS contains a static switch that will automatically transfer loads to a bypass power source in the event of either UPS failure or a downstream fault. The bypass power source is obtained from the same 600 VAC power panel as the UPS. Bypass power is conditioned by a step-down transformer with no load taps which permits manual compensation for large variations in source voltage.

References Provided: NONE

Question 49 Cross References (table item links)

10CFR55

- 41(b)(5)

Cognitive Level

- 1

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 262002 A2.01 2.6/2.8 Under voltage
A2.01 Ability to (a) predict the impacts of the following on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Under voltage.

Question Source

- New

PROC

- N1-OP-40 Rev. NA

Question Setting

- C1 (License class closed reference)

50 **SYSID: 21050** **Points: 1.00**

Following a loss of Battery Board 11, which one of the following loads affected by the power loss can be returned to operation after aligning alternate 125 VDC power to Battery Board 12?

A. All functions of ERV 111, 112, and 113 are restored.

B. All functions of PB 101 breakers R1011 and R1014.

C. EDG 102 except breaker R1022 must be manually closed.

D. Motor Driven Fire Pump except its auto start remains defeated.

Answer: C

Associated objective(s):

Development Area (FIO)

Question 50 Details

Question Type:	Multiple Choice
Topic:	NRC RO 50
System ID:	21050
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-263-1-01, EO-1.8

Answer: c. EDG 102 125 VDC control power is lost.
Transfer to alternate 125 VDC supply for EDG102 per N1-OP-45 Sec. H.

3.0 <u>EDG Alternate 125 VDC Feeds</u>

3.1 To supply alternate 125 VDC feed to EDG 102, perform the following:

3.1.1 Manually close DC ISOLATION BKR. CAB. 103 - BATT 12 TO 102 DG (DG 103 RM) .... (_)

3.1.2 Select to BATT 12, UNIT-SEL. SW. (DG 102 Control Cab) .... (_)

3.1.3 IF EDG start and loading is desired in this configuration,
THEN local manual closure of R1022 DG 102 OUTPUT BKR is required .... (_)

Distractor: a. The following functions for ERV 111, 112 & 113 are NOT operable:
- F Panel controls
- Auto Pressure Relief
- ADS

Distractor: b. Powered from BB12 and are not affected.

Distractor: d. Diesel Fire Pump control circuits are DC powered, not the MDFP. MDFP is not affected by BB11 loss. Indicating auto start feature remaining inoperable is similar to the actual affected control circuits of the DFP.

References Provided: NONE

Question 50 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 1

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 263000 K2.01 3.1/3.4 Major DC loads, Rev. NA
K2.01 Knowledge of electrical power supplies to the following: Major D.C. loads

Question Source

- New

PROC

- N1-OP-47A Rev. NA

Question Setting

- C1 (License class closed reference)

51 **SYSID: 21051** **Points: 1.00**

Given the following conditions:

- A LOCA occurred two (2) minutes ago
- Drywell pressure is 14 psig and lowering
- RPV water level is -30 inches and rising
- Containment Spray Pumps 112 and 122 were manually started for containment spray
- Containment Spray Pumps 111 and 121 are locked out

Subsequently, a loss of offsite power occurs. EDG102 starts. EDG103 fails to start and cannot be manually started.

Which one of the following describes the Containment Spray Pump(s) running one (1) minute after the power loss?

A. Only 112.

B. 111 and 112. Typo Not plausible since locked out

C. Only 122.

D. 112 and 122.

Answer: A

Associated objective(s):

Development Area (FIO)

Question 51 Details

Question Type:	Multiple Choice
Topic:	NRC RO 51
System ID:	21051
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-264-1-01, EO-1.8

Answer: a. Containment Spray Pump 112 will restart 30 seconds after EDG102 powers PB102. Containment Spray Pump 122 will not start since it is powered from PB103 which is supplied by EDG103 which cannot power the bus.

Note: Answers are plausible because the Core Spray pumps with the same number (i.e., 111) are not all powered from the same power boards as the Containment Spray Pumps.

38.0 De-energize/Re-energize Power Board 103

NOTES: 1. The SSS may NA steps in the following subsection based on existing plant conditions.

2. PB 17A and PB 17B will be cross tied per Section H.27.0.

38.1 IF de-energizing PB 103, THEN perform the following:

38.1.1 SSS has reviewed loads powered from PB 103 for effects on present plant conditions . . . . (_)

38.1.2 Cross tie PB 17B with PB 17A per Section H.27.0 . (_)

38.1.3 Install fuse FU 12 at PB 103 Pot Trans Cube in OFF (_)

38.1.4 Isolate AND apply Clearance Tags to EDG 103 starting air . . . . (_)

38.1.5 Verify the following control switches in PULL TO LOCK:

- CORE SPRAY PUMP 122 . . . . (_)
- CORE SPRAY TOPPING PUMP 122 . . . . (_)
- CONTAINMENT SPRAY RAW WATER PUMP 122 . . . . (_)
- CONTAINMENT SPRAY PUMP 122 . . . . (_)
- CONTAINMENT SPRAY RAW WATER PUMP 121 . . . . (_)
- CONTAINMENT SPRAY PUMP 121 . . . . (_)
- CORE SPRAY PUMP 112 . . . . (_)
- CORE SPRAY TOPPING PUMP 112 . . . . (_)

Distractor: b. Containment Spray Pump 112 will restart 30 seconds after EDG102 powers PB102. Containment Spray Pump 111 will not start even though it is powered from PB102 it is locked out meaning it is in Pull-To-Lock and wont start even though a start signal is received.

Distractor: c. Containment Spray Pump 122 will not start since it is powered from PB103 which is supplied by EDG103 which cannot power the bus.

Distractor: d. Containment Spray Pump 122 will not start since it is powered from PB103 which is supplied by EDG103 which cannot power the bus. Containment Spray Pump 112 will not start because PB103 has no power and it is locked out meaning it is in Pull-To-Lock and wont start even though a start signal is received.

References Provided: NONE

Question 51 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 1

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 264000 K3.03 4.1*/4.2* Major loads powered from electrical buses fed by the emergency generator(s) K3.03 Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on following: Major loads powered from electrical buses fed by the emergency generator(s).

Question Source

- New

PROC

- N1-OP-30 Rev. NA

Question Setting

- C1 (License class closed reference)

52 **SYSID: 21052** **Points: 1.00**

The plant is at 100% power with the following:

- Instrument Air Compressor (IAC) #13 in service
- IAC #12 in pull-to-lock
- Instrument Air Dryer (IAD) 94-168 in service
- Subsequently, IAC #11 trips (lost control power) and its control switch is placed in pull-to-lock
- The ASSS directs bypass of IAD 94-168 and IAD 94-169

Per N1-OP-20, which one of the following actions is required until either IAC #11 or IAC #12 is returned to service?

A. Start IAD 94-169 and then close the bypass valves.

B. Blow down designated air manifolds once every 24 hours.

C. Align the temporary service air compressor to the instrument air system.

D. Align service air to the reactor building track bay roll door (D-39) inflatable seal.

Answer: B

Associated objective(s):

Development Area (FIO)

Question 52 Details

Question Type:	Multiple Choice
Topic:	NRC RO 52
System ID:	21052
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-278-1-01, EO-1.8

N1-OP-20, D.4.0, H.3.0 step 3.6

Answer: b. Per N1-OP-20; D.4.0: Any combination of pulling control power fuses AND/OR placing Control Switches to Pull To Lock for both Instrument Air Compressors 11 AND 12 will result in a Loss of Control Power to the Instrument Air Dryers 94-168 and 94-169, resulting in a shutdown of the Instrument Air Dryers. Per N1-OP-20, Section H.3.0, step 3.6, blow down designated air manifolds daily until air dryers are restored.

Distractor: a. Per N1-OP-20; D.4.0: Any combination of pulling control power fuses AND/OR placing Control Switches to Pull To Lock for both Instrument Air Compressors 11 AND 12 will result in a Loss of Control Power to the Instrument Air Dryers 94-168 and 94-169, resulting in a shutdown of the Instrument Air Dryers. Neither IAC can be started at this time.

Distractor: c. The Temporary Service Air Compressor is not an acceptable alternate supply to the Instrument Air System. The temporary air compressor is used during outages for service air augmentation.

Distractor: d. Per N1-OP-20; H.17.0 note: The Reactor Building Track Bay Roll Door (D-39) inflatable seal is pressurized from the Instrument Air System and cannot be considered operable with IAC 11 and IAC 12 removed from service. Therefore, failure to maintain

the Rx Bldg Outer Swing Door (D-198) closed and sealed while IAC 11 and IAC 12 are removed from service will result in violation of secondary containment integrity.
Per N1-OP-20; H.17.0, Step 17.4: If secondary containment integrity is required, THEN verify the following: (1) Rx Bldg Outer Swing Door (D-198) is closed and sealed, and (2) Clearance section placed on Rx Bldg Outer Swing Door (D-198) in the closed and sealed position.

References Provided: NONE

Question 52 Cross References (table item links)

10CFR55
- 41(b)(5)

Cognitive Level
- 2

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- 300000 A2.01 2.9/2.8 Air dryer and filter malfunctions
A2.01 Ability to (a) predict the impacts of the following on the INSTRUMENT AIR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Air dryer and filter malfunctions.

Question Source
- New

PROC
- N1-OP-20 Rev. NA

Question Setting
- C1 (License class closed reference)

53 **SYSID: 21053** **Points: 1.00**

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

- The SW PUMP HEADER PRESS alarm is being intermittently received
- 11 SW Pump current is observed oscillating between 20-70 AMPS

Which one of the following will cause the above alarm and indication and include the appropriate action?

A. 11 SW Pump cavitaton/start 12 SW Pump

B. 12 SW Pump check valve leakage/close valve 72-75, SW Pump discharge cross connect

C. 11 Adams Strainer clogging/place the backwash timer in MANUAL

D. 11 Adams Strainer backwash valve leakage/place the backwash timers in MANUAL

Answer: A

Associated objective(s):

Development Area (FIO)

Question 53 Details

Question Type:	Multiple Choice
Topic:	NRC RO 53
System ID:	21053
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-276-1-01, EO-1.8

This question was derived from bank question SYSID 15656

Answer: a. Correct. Oscillating amps is an indication pump cavitation.

Distractor: b, c, d Incorrect. Check valve leakage and Adams Strainer malfunctions will not result in oscillating amps.

References Provided: NONE

Question 53 Cross References (table item links)

10CFR55
- 41(b)(7)

Cognitive Level
- 1

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- 400000 A4.01 3.1/3 CCW indications and control
Ability to manually operate and/or monitor the following from the control room:
CCW indications and control

Question Source
- Bank

PROC
- N1-ARP-H1 Rev. NA
- N1-OP-18 Rev. NA

Question Setting
- C1 (License class closed reference)

54 **SYSID: 21054** **Points: 1.00**

A reactor startup is in progress, with the following:

- IRMs have been placed on Range 7
- Annunciator F3-1-5, CRD CHARGING WTR HEADER PRESSURE HI/LO has been received
- RD19, Charging Header Pressure on F panel indicates 1550 psig
- A reactor scram occurs

Which one of the following describes the effect of this condition on control rod drive mechanisms?

A. Slower than design insertion speed

B. May Result in Damage to collet fingers

C. May Result in Damage to index tube

D. Faster than design insertion speed

Answer: C

Associated objective(s):

Development Area (FIO)

Question 54 Details

Question Type:	Multiple Choice
Topic:	NRC RO 54
System ID:	21054
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-201-1-01, EO-1.8

This question was derived from bank SYSID 16223
EO-1.6
EO-1.8

The student must recognize that under the conditions presented, lower than rated RPV pressure exists. Per N1-OP-5, a scram with elevated charging header pressure (above 1510 psig) when the reactor is not at rated conditions may result in damage to the mechanism index tube. This damage is due to the excessive differential pressure across the tube and not the velocity (speed) of the rod during insertion. This question requires the student to recall the basis for the CRD Charging Water Header pressure upper limitations after discerning the applicable conditions for the precaution.

"...collect finger damage..." is incorrect. As stated above. Selection of this answer indicates the student is unfamiliar with the components of the mechanism and the basis for limits and alarm response actions.

"...slower scram insertion speed..." is incorrect. This is a concern only if RPV pressure is less than 800 psig and a complete loss of all accumulator pressure is experienced. The normal pressure required for the HCU accumulators prevents this.

"...faster scram insertion speed..." is incorrect. Higher pressures (1600 psig charging header with 0 psig RPV) have resulted in deformation of the index tube when control rods have short withdrawal (12 inches) but do not appreciably affect the scram velocity (testing has been done at velocities up to 100 inches per second).
References Provided: NONE

Question 54 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 2

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 201001 K3.03 3.1/3.2 Control rod drive mechanisms
K3.03 Knowledge of the effect that a loss or malfunction of the CONTROL ROD DRIVE HYDRAULIC SYSTEM will have on following: control rod drive mechanisms.

Question Source

- Bank

PROC

- N1-ARP-F3 Rev. NA
- N1-OP-05 Rev. NA
- N1-ICP-A-44, Rev. NA

Question Setting

- C1 (License class closed reference)

55 **SYSID: 21055** **Points: 1.00**

The operator is performing N1-ST-V3, RWM Operability Test, Section 8.4, RWM Rod Block Test, in preparation for reactor startup. An "out of sequence" control rod is selected (SELECT ERROR light ON) and withdrawn to position 04. The following alarms are received:

- WITHDRAW BLOCK light ON at RWM Display Panel
- ROD WORTH MINIMIZER light ON at Rod Block Monitor Display Panel
- F3-4-4, ROD BLOCK in alarm

Which one of the following describes the operation of the RWM?

A. Rod block alarms are an accurate representation of correct operation.

B. Rod block alarms should have been received when the rod was selected.

C. Rod block alarms should have been received when attempting to withdraw this control rod from position 00.

D. Rod block alarms should not be received until attempting to withdraw this control rod to position 06.

Answer: ~~A~~ D

Associated objective(s):

Development Area (FIO)

Question 55 Details

Question Type:	Multiple Choice
Topic:	NRC RO 55
System ID:	21055
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-201-1-03, EO-1.7

Answer: d. 8.4 RWM Rod Block Test
8.4.1 Select an "out of sequence" Rod at position 00 as designated by Reactor Engineering AND record Rod Selected.
8.4.2 Confirm SELECT ERROR light ON at Rod Worth Minimizer Display Panel.
8.4.3 Using CONTROL ROD MOVEMENT Switch in ROD OUT NOTCH mode, withdraw designate Control Rod to "04" position. 8.4.4 Verify Rod motion to position 04.
8.4.5 Using CONTROL ROD MOVEMENT Switch in ROD OUT NOTCH [T/S] mode, attempt to withdraw designated Control Rod to "06" position AND confirm Rod motion is blocked beyond position "04" is blocked.
8.4.6 Confirm the following indications ON: (T/S)
a. WITHDRAW BLOCK light ON at RWM Display Panel.
b. ROD WORTH MINIMIZER light ON at Rod Block Monitor Display Panel.
c. Annunciator F3-4-4, ROD BLOCK - ON.

Distractor: a. b. c. See justification above.

References Provided: None

Question 55 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 2

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 201006 A3.03 3.1/3 Annunciator and alarm signals: P-Spec(Not-BWR6)
A3.03 Ability to monitor automatic operations of the ROD WORTH MINIMIZER SYSTEM (RWM) including: Annunciator and alarm signals.

Question Source

- New

PROC

- N1-ST-V3 Rev. NA

Question Setting

- C1 (License class closed reference)

56 **SYSID: 21056** **Points: 1.00**

The plant has scrammed and all the control rods have NOT fully inserted. EOP-3, FAILURE TO SCRAM has been entered. The following conditions exist:

- Main Turbine has tripped
- TBVs are controlling reactor pressure
- RPV water level is being lowered only CRD is injecting
- Reactor power is 8%

Which one of the following actions is required at this time?

A. Initiate boron injection

B. Trip the recirculation pumps

C. Initiate the Emergency Condensers

D. Lower RPV water level to -84 inches

Answer: B

Associated objective(s):

Development Area (FIO)

Question 56 Details

Question Type:	Multiple Choice
Topic:	NRC RO 56
System ID:	21056
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-344-1-03, EO-1.2

Answer:	b. Trip the recirculation pumps per EOP-3 because the turbine has tripped.
Distractor:	a. Boron injection is not required at this time because power is NOT oscillating and there is no heat input to the torus.
Distractor:	c. No ERVs are cycling therefore the Ecs are NOT required.
Distractor:	d. RPV water level should be lowered to -41 inches NOT -84 inches because the ERVs are closed.

References Provided: ALL EOPs

Question 56 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 2

Difficulty Level
- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- G2.4.6 3.1/4 Knowledge symptom based EOP mitigation strategies
- 202002 Recirculation Flow Control, Rev. NA

Question Source
- New

PROC
- N1-EOP-3 Rev. NA

Question Setting
- C1 (License class closed reference)

57 **SYSID: 21057** **Points: 1.00**

During a control rod sequence exchange at power, the first control rod to be repositioned is at position 08 and will be single notch withdrawn to position 48. This control rod is successfully withdrawn to position 10 but when withdrawn to position 12, the Rod Position Indication (RPI) for this control rod is blank.

Per N1-OP-5, Control Rod Drive System, abnormal for Loss of RPI, which one of the following actions is required to proceed ~~if you suspect a failed reed switch for position 12~~?

A. Insert this control rod one notch and confirm rod position by observing proper RPI for position 10.

B. Declare this control rod inoperable and without delay fully insert this control rod and then disarm and isolate its' HCU

C. Return this control rod to its original position 08 and request Reactor Engineering modify the control rod sequence exchange.

D. Insert this control to rod full in and confirm rod position by observing the green full in light is on.

Answer: A

Associated objective(s):

Development Area (FIO)

Question 57 Details

Question Type:	Multiple Choice
Topic:	NRC RO 57
System ID:	21057
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-201-1-01, EO-1.7

Answer:	a. If the control rod is inserted one notch and position can be determined, then control rod withdrawal can proceed
Distractor:	b. This is the required action if the position indication is lost for more than one position. This has not occurred – only one position indication is lost
Distractor:	c. The control rod sequence exchange can proceed without any modifications. An appropriate action for any abnormal condition is to place the component in a safe condition. Inserting this control rod back to position 08 would occur without any procedural guidance which is not appropriate.
Distractor:	d. Numerous control rod failures require that the affected control rod be fully inserted. This is an incorrect application of this procedure to the loss of RPI actions. It is only necessary to insert the control rod one notch to confirm its position.

References Provided: None

Question 57 Cross References (table item links)

<u>10CFR55</u>

- 41(b)(5)

<u>Cognitive Level</u>

- 1

<u>Difficulty Level</u>

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

<u>NUREG 1123 KA Catalog Rev. 2</u>

- 214000 A2.01 3.1/3.3 Failed reed switches
A2.01 Ability to (a) predict the impacts of the following on the ROD POSITION INFORMATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
Failed reed switches.

<u>Question Source</u>

- New

<u>PROC</u>

- N1-OP-5 Rev. NA

<u>Question Setting</u>

- C1 (License class closed reference)

58 **SYSID: 21058** **Points: 1.00**

With EC 11 aligned for standby per the operating procedure, which one of the following conditions will automatically initiate EC 11 and EC 12?

A. Steam leak in the drywell estimated at 342 gpm.

B. Coolant leak in the drywell estimated at 342 gpm.

C. RPV water level is deliberately lowered during an ATWS.

D. RPV pressure is controlled by bypass valves during an ATWS.

Answer: C

Associated objective(s):

Development Area (FIO)

Question 58 Details

Question Type:	Multiple Choice
Topic:	NRC RO 58
System ID:	21058
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-207-1-01, EO-1.4

Answer: c. Automatic operation of the emergency cooling system is initiated by high reactor pressure, in excess of 1080 psig, sustained for 12 seconds. To assist in depressurization for small breaks, the system is initiated on low-low reactor water level (+5 inches) sustained for 12 seconds.
If reactor water level is deliberately lowered during an ATWS, it is lowered to at least -41 inches. This satisfies the conditions for EC initiation on lo-lo level.

Distractor: a. High reactor pressure is not challenged, but rather lowers in response to the leak. Reactor water level remains above the lo-lo set point if within the capability of HPCI.
During a loss of coolant accident, high drywell pressure due to a line break would cause a reactor scram. The automatic scram will cause a turbine trip after a five second delay. The shaft-driven feedwater pump will provide feedwater flow of greater than 3800 gpm for approximately 3.2 minutes during pump coastdown. The turbine trip will signal the motor-driven feedwater pump to start. This signal will be simultaneous with the start of the shaft-driven pump coastdown. The motor-driven feedwater pump will be up to speed and capable of supplying 3420 gpm in about 10 seconds. As a backup to the turbine trip signal, low reactor water level (53") will also signal the motordriven pump to start. This

ensures a continuous, uninterrupted supply of high pressure feedwater to the reactor. The electric motor-driven condensate pumps and feedwater booster pumps are capable of providing 3420 gpm at reactor pressures below approximately 332 psig. The flow control valve would admit the full flow of 3420 gpm into the reactor until the low reactor level was regained. At this point, the flow control valve would reduce flow into the reactor until the flow matched the flow out of the break. Necessary pump recirculation to accommodate the decreasing system flow has been provided as a part of the basic pump control system. Consequently, the pump is capable of delivering any flow into the reactor from 3420 gpm down to zero flow.

Distractor: b. High reactor pressure is not challenged, but pressure will remain closer to normal as compared to the steam break. Reactor water level remains above the lo-lo set point if within the capability of HPCI. See discussion for "a" above.

Distractor: d. If the main condenser is available as a heat sink, high reactor pressure is not a concern. If the main condenser is lost as a heat sink during an ATWS, then reactor pressure would likely go high (depending on the reactor power level) resulting in initiation of the ECs if ERV pressure control is not established in a timely manner.

References Provided: NONE

Question 58 Cross References (table item links)

10CFR55
- 41(b)(7)

Cognitive Level
- 2

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- 216000 K4.05 3.9/4.1 Initiation of the emergency core cooling systems
Knowledge of NUCLEAR BOILER INSTRUMENTATION design feature(s) and/or interlocks which provide for the following — Initiation of ECCS

Question Source
- New

PROC
- N1-OP-13 Rev. NA

Question Setting
- C1 (License class closed reference)

59 **SYSID: 21059** **Points: 1.00**

The plant is at 100% power. Because of a leaking ERV, torus cooling is in service per N1-OP-14; H.5.0, Cooling Torus Water Temperature Using 111 Containment Spray System. When the desired Torus water temperature is reached, the following actions are performed at the specified times:

08:00 Containment Spray Pump 111 and Containment Spray Raw Water Pump 111 secured
09:30 Draining of the Containment Spray Raw Water Heat Exchanger shell side is complete
09:35 80-16, CONT SPRAY DISCH IV 111, is opened
09:40 80-40, CONT SPRAY BYPASS BV 11, is opened

For these conditions, which one of the following is the EARLIEST TIME at which the containment spray system can be declared operable?

A. 08:00

B. 09:30

C. 09:35

D. 09:40

Answer: D

Associated objective(s):

Development Area (FIO)

Question 59 Details

Question Type:	Multiple Choice
Topic:	NRC RO 59
System ID:	21059
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-226-1-01, EO-1.7

Answer: d. The shell side (containment spray) side of the system must be drained and then the IVs opened to return to operable status. Opening 80-16 restores operability for some containment spray but 80-40 must also be opened (Appendix J water seal). (See N1-OP-14, D.11.0)

Distractor: a. Securing the pumps places the system in standby (no pumps running) but additional actions are required to restore complete operability.

Distractor: b. The shell side (containment spray) side of the system must be drained and then the IVs opened to return to operable status. If the IV were opened before draining water would flow into the drywell.

Distractor: c. Opening 80-16 restores operability for some containment spray but 80-40 must also be opened (Appendix J water seal).

References Provided: NONE

Question 59 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 2

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- G2.1.23 3.9/4 Ability to perform specific system and integrated plant procedures during different modes of plant operation

Question Source

- New

PROC

- N1-OP-14 Rev. NA

Question Setting

- C1 (License class closed reference)

60 **SYSID: 21060** **Points: 1.00**

Following full power operation for one hundred (100) days, Offgas and Main Steam Line (MSL) radiation levels have been rising. Radiation levels in relation to the Normal Full Power Background (NFPB) are confirmed as follows

- MSL Radiation Monitor CH 11 is 3.9 times the NFPB
- MSL Radiation Monitor CH 12 is 4.1 times the NFPB

Which one of the following describes the required action(s) in response to the above conditions?

A. Start the Control Room Emergency Ventilation System in the emergency recirculation mode.

B. Start the Reactor Building Emergency Ventilation System in the filtration mode and isolate the reactor building.

C. Insert a manual reactor scram and then place the vessel isolation 11 and 12 control switches to the isolation position.

D. Reduce reactor power to 60% to reduce MSL radiation levels below their current levels until the fuel leak can be suppressed.

Answer: C

Associated objective(s):

Development Area (FIO)

Question 60 Details

Question Type:	Multiple Choice
Topic:	NRC RO 60
System ID:	21060
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-239-1-01, EO-1.7

Answer: c. For Upscale Alarm: IF Main Steam Radiation Monitor 11 reaches 3.75XNFPB AND channel 12 MSLRM at 3.75 x NFPB, THEN perform the following:
a. Initiate a Manual Reactor Scram.
b. Initiate a Manual Vessel Isolation.

Distractor: a. If indications of a Steam Line Break exist, then verify CREVS initiated. These are indications of fuel failure. Also, In the event the following parameters/actions take place, manual initiation of EVS is required: (1) Total iodine radioactivity concentration >9.47 Ci/gm (not known at this time) or Unit 2 enters RR (Radioactivity Release) EOP.
Also, In the event that any of the following signals are received:
(1) Outside air contamination of 168 cpm above background on radiation monitors, or (2) LOCA (Lo-Lo level or High Drywell Pressure) or (3) MSLB (Hi steam tunnel temperature OR high steam flow) will initiate CREVS.

Distractor: b. Starting RBEVS and isolating the reactor building is appropriate if secondary containment is affected. For these conditions secondary containment is not the concern. The fuel leak has resulted in elevated radiation levels in the steam lines which affect the turbine building areas rather than the reactor building.

Distractor: d. When the fuel leak is suspected suppression testing would be performed and the affected fuel assembly or assemblies suppressed to reduce the magnitude of the leak. This would occur much earlier than now. Based on the magnitude of the fuel leak, a reactor scram and vessel isolation are required.

References Provided: NONE

Question 60 Cross References (table item links)

10CFR55

- 41(b)(5)

Cognitive Level

- 2

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 239001 A2.05 3.9/4.2 Main steam line high radiation
A2.05 Ability to (a) predict the impacts of the following on the MAIN AND REHEAT STEAM SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
Main steam line high radiation.

Question Source

- New

PROC

- N1-ARP-F1 Rev. NA
- N1-ARP-F4 Rev. NA

Question Setting

- C1 (License class closed reference)

61 **SYSID: 21061** **Points: 1.00**

The plant is operating at 90% power, with the following:

- Power ascension is in progress
- Turbine Load Limit is set at 95%

As reactor power is raised, which one of the following effects will be seen when the Turbine Load Limit setpoint is reached?

A. Reactor scram on high pressure.

B. TURBINE BYPASS VALVES OPEN alarm.

C. TURBINE AUTOMATIC GOVERNOR RUNBACK alarm.

D. Reactor power oscillations.

Answer: B

Associated objective(s):

Development Area (FIO)

Question 61 Details

Question Type:	Multiple Choice
Topic:	NRC RO 61
System ID:	21061
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OP-001-248-1-01, EO-1.4

This question derived from bank SYSID 17947
EO-1.4.b

Answer:	b is correct – Pressure is controlled by control, then bypass valves. Load is controlled by control valves. The Load Limiter may be employed to limit control valve opening (Load).
Distractor:	a is incorrect – Ignores operation of Bypass valves to control reactor pressure.
Distractor:	c. Alarm would be received in response to stator water cooling runbacks signal, not reaching load limit.
Distractor:	d. TBVs open to control reactor pressure. Conditions for power oscillations are not present.

References Provided: NONE

Question 61 Cross References (table item links)

10CFR55

- 41(b)(5)

Cognitive Level

- 2

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 241000 A1.15 3.1/3.1 Maximum combined flow limit
A1.15 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR/TURBINE PRESSURE REGULATING SYSTEM controls including:
Maximum combined flow limit.

Question Source

- New

PROC

- N1-OP-31 Rev. NA

Question Setting

- C1 (License class closed reference)

62 **SYSID: 21062** **Points: 1.00**

Which one of the following describes the potential consequences, if any, of operating with Feedwater Flow Control Valves 11 and 12 in automatic with feed flow at 2 x 10^6 lbs/hr per N1-OP-16, Feedwater System Booster Pump to Reactor?

A. More than one flow control valve in auto may cause unstable system performance.

B. One flow control valve could go full open causing pump runout and motor damage.

C. One flow control valve could go full closed causing water hammer and pipe whip.

D. There are no potential consequences when operating in this configuration.

Answer: A

Associated objective(s):

Development Area (FIO)

Question 62 Details

Question Type:	Multiple Choice
Topic:	NRC RO 62
System ID:	21062
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-259-1-01, EO-1.7 This question was derived from bank SYSID 12567 **References Provided: NONE**

Question 62 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 1

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 259001 A4.04 3.1/2.9 System valves
A.4.04 Ability to manually operate and/or monitor in the control room: System Valves

Question Source

- Bank

PROC

- N1-OP-16, Rev. NA

Question Setting

- C1 (License class closed reference)

63 **SYSID: 21063** **Points: 1.00**

The plant is operating at 100% power. A loss of RPS Bus 12 occurs.

Which one of the following describes the response of the Reactor Building (RB) Ventilation System and Reactor Building Emergency Ventilation (RBEVS)?

A. RB Ventilation isolates. RBEVS is running.

B. RB Ventilation continues to operate. RBEVS is running.

C. RB Ventilation isolates. RBEVS is off but a ½ actuation signal is present.

D. RB Ventilation continues to operate. RBEVS is off but a ½ actuation signal is present

Answer: A

Associated objective(s):

Development Area (FIO)

Question 63 Details

Question Type:	Multiple Choice
Topic:	NRC RO 63
System ID:	21063
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-273-1-01, EO-1.5

Answer: a. Although power is lost to only one of the two radiation monitors, which would not in itself cause a RB isolation and RBEVS start, the loss of power to the isolation/actuation logic affects both instruments within the isolation/actuation logic causing the RB isolation and RBEVS start.

Distractor: b,c,d, all are incorrect because either logic channel will cause the other logic channel to trip.

References Provided: NONE

Question 63 Cross References (table item links)

10CFR55
- 41(b)(7)

Cognitive Level
- 1

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- 272000 K6.01 3/3.2 Reactor protection system
K6.01 Knowledge of the effect that a loss or malfunction of the following will have on the RADIATION MONITORING SYSTEM: Reactor Protection System.

Question Source
- New

PROC
- N1-OP-40 Rev. NA

Question Setting
- C1 (License class closed reference)

64 **SYSID: 21064** **Points: 1.00**

The Reactor Building (RB) differential pressure (d/p) controller has been placed in manual per N1-OP-10, H.7.0, Controller Operation in Manual

- The operator pressed the **A/M** button to change from auto to manual control and confirmed the controller displayed an **M**
- The operator confirmed damper control selected.
- With reactor building d/p at -0.30 psid, the operator presses the controller up arrow (▲) momentarily.

Which one of the following describes the response of the RB ventilation system and the resultant change in RB d/p?

A. Raise supply flow and degrade the RB d/p (less negative).

B. Lower supply flow and improve the RB d/p (more negative).

C. Raise supply flow and improve the RB d/p (more negative).

D. Lower supply flow and degrade the RB d/p (less negative).

Answer: A

Associated objective(s):

Development Area (FIO)

Question 64 Details

Question Type:	Multiple Choice
Topic:	NRC RO 64
System ID:	21064
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-288-1-01, EO-1.7

Answer:	a. Increasing output/damper position will increase (raise) supply flow and decrease (lower) reactor building differential pressure (less negative). Decreasing output/damper position will decrease (lower) supply flow and increase building differential pressure (more negative).
Distractor:	b. Supply flow raises and d/p degrades.
Distractor:	c. With supply flow raising the d/p degrades.
Distractor:	d. Supply flow raises not lowers. With supply flow lowering the d/p improves

References Provided: NONE

Question 64 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 2

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 288000 K5.02 3.2/3.4 Differential pressure control
K5.02 Knowledge of the operational implications of the following concepts as they apply to PLANT VENTILATION SYSTEMS: Differential pressure control.

Question Source

- New

PROC

- N1-OP-10 Rev. NA

Question Setting

- C1 (License class closed reference)

65 **SYSID: 21065** **Points: 1.00**

During the month of MAY, the Control Room Ventilation System chiller alignment is changed as follows:

- Locally at 210.1-119, CONTROL ROOM CHILLER 11, the operator depresses pushbutton Compressor Sequence 1
- Locally at 210.1-120, CONTROL ROOM CHILLER 12, the operator depresses pushbutton Compressor Sequence 1

Statements is Correct

Per N1-OP-49, Control Room Ventilation System, which one of the following ~~describes the accurateness of t~~he operator actions?

regarding

A. Mis-operation aligning 60% compressors 111 and 121 for LAG operation.

B. Mis-operation aligning 40% compressors 112 and 122 for LAG operation.

C. Correct operation aligning 60% compressors 111 and 121 for LEAD operation.

D. Correct operation aligning 40% compressors 112 and 122 for LEAD operation.

Answer: C

Associated objective(s):

Development Area (FIO)

Question 65 Details

Question Type:	Multiple Choice
Topic:	NRC RO 65
System ID:	21065
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-288-1-03, EO-1.7

Answer: c. The Control Room Ventilation System chillers are operated with the 60% compressors (111 and 121) in lead during warm months (May - September), and the 40% (112 and 122) compressors in lead during the cool months (November - March). This alignment minimizes unnecessary cycling of the compressors.

7.0 Shifting Chiller Unit Lead Compressors From 40% to 60%
NOTE: This section is used **every spring** to shift lead compressor alignment for summer operation.
7.1 Locally at 210.1-119, CONTROL ROOM CHILLER 11, depress pushbutton **Compressor Sequence 1**
7.2 Locally at 210.1-120, CONTROL ROOM CHILLER 12, depress pushbutton **Compressor Sequence 1**
8.0 Shifting Chiller Unit Lead Compressors From 60% to 40%
NOTE: This section is used every **autumn** to shift lead compressor alignment for winter operation.
8.1 Locally at 210.1-119, CONTROL ROOM CHILLER 11, depress pushbutton **Compressor Sequence 2**
8.2 Locally at 210.1-120, CONTROL ROOM CHILLER 12, depress pushbutton **Compressor Sequence 2**

Distractor: a. Compressors 111 and 121 are aligned for lead operation which is a correct operation for the month of May.

Distractor: b. Compressors 112 and 122 are aligned for lag operation however this is a correct operation for the month of May, not a mis-operation.

Distractor: d. Compressors 111 and 121 are aligned for lead operation not compressors 112 and 122. Compressors 112 and 122 aligned for lead operation is correct for the months of November through March.

References Provided: NONE

Question 65 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 2

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 290003 A3.01 3.3/3.5 Initiation/reconfiguration
A3.01 Ability to monitor automatic operations of the CONTROL ROOM HVAC including: Initiation/reconfiguration.

Question Source

- New

PROC

- N1-OP-49 Rev. NA

Question Setting

- C1 (License class closed reference)

66 **SYSID: 21066** **Points: 1.00**

Following seven (7) days of vacation, an operator works twelve (12) hours on the first day back on shift and then works an additional four (4) hours of overtime.

Assume NO extension is authorized, which one of the following describes the MAXIMUM number of hours this operator can work the next day WITHOUT exceeding the Nine Mile Point working hour limitations?

A. 4 hours

B. 8 hours

C. 12 hours

D. 16 hours

Answer: B

Associated objective(s):

Development Area (FIO)

Question 66 Details

Question Type:	Multiple Choice	
Topic:	NRC RO 66	
System ID:	21066	
User ID:		
Status:	Active	
Must Appear:	No	
Difficulty:	0.00	
Time to Complete:	0	
Point Value:	1.00	
Cross Reference:	LC1 03-01	
User Text:		
User Number 1:	0.00	
User Number 2:	0.00	
Comment:	Answer:	b. The restrictions are 16 hours in any 24-hour period and 24 hours in any 48-hour period. Since the operator worked 16 hours on the first day, then without any extension the operator is restricted to 8 hours on the second day of work.
	Distractor:	a. Can work 8 hours
	Distractor:	c. Can only work 8 hours without an approved extension
	Distractor:	d. Can only work 8 hours without an approved extension
	References Provided: NONE	

Question 66 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 2

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- G2.1.10 2.7/3.9 Knowledge of conditions and limitations in the facility license

Question Source
- New

Question Setting
- C1 (License class closed reference)

67 **SYSID: 21067** **Points: 1.00**

Regarding cardox valves **EMPC 99-92A** and **HBV 99-92** for ZONE C-2141, EDG 102, and assuming the cardox system responds per design, which one of the following describes their integrated valve operation in response to a valid fire detection system actuation for this zone?

A. EMPC 99-92A is <u>energized</u> to supply pneumatic pressure, which opens HBV 99-92.

B. EMPC 99-92A is <u>energized</u> and vents pneumatic pressure, which allows HBV 99-92 to open.

C. EMPC 99-92A is <u>de-energized</u> to supply pneumatic pressure, which opens HBV 99-92.

D. EMPC 99-92A is <u>de-energized</u> and vents pneumatic pressure, which allows HBV 99-92 to open.

Answer: A

Associated objective(s):

Development Area (FIO)

Question 67 Details

Question Type:	Multiple Choice
Topic:	NRC RO 67
System ID:	21067
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Answer: a. EMPC- 99-92A is normally de-energized and must be energized. When energized it repositions to supply cardox (pneumatics) to open HBV 99-92. The same valid actuation will cause normally energized EMPC 99-97A to deenergize and supply cardox (pneumatics) to open HBV-99-97.

Distractors: b. c. d. See justification above.

References Provided: The following prints are provided for use when answering this question:

- C-18000-C, Piping, Instrument, and Equipment Symbols
- C-18039-C, Cardox Fire Extinguishing System P&I Diagram

Question 67 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 2

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

DRW
- C-18000-C Rev. na
- C-18039-C Rev. NA

NUREG 1123 KA Catalog Rev. 2
- G2.1.24 2.8/3.1 Ability to obtain and interpret station electrical and mechanical drawings

Question Source
- New

Question Setting
- C1 (License class closed reference)

68 **SYSID: 21068** **Points: 1.00**

The plant is preparing to perform a startup with the following conditions:

- You are performing valve lineup of the Backfill System
- Valve 28.1-43 has a Caution Clearance Section tag attached to the valve
- The tag indicates the valve is to remain closed until Backfill is ready to be initiated

Which one of the following describes the actions required to complete this valve lineup?

The operator shall .....

A. leave the valve in its present position, notify the CSO and SSS of the discrepancy and note the discrepancy

B. leave the valve in its present position and indicate the Clearance Section Number in the INITIALS/DATE column

C. remove the Clearance Section tag, indicate the Clearance Section number on the valve lineup and position the valve

D. remove the Clearance Section tag and update the Clearance Section Tag Removal Sheet then position the valve

Answer: B

Associated objective(s):

Development Area (FIO)

Question 68 Details

Question Type:	Multiple Choice
Topic:	NRC RO 68
System ID:	21068
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	This question derived from bank SYSID 17196

Answer: b. indicate Clearance Section Number in INITIALS/DATE is **correct** - Per N1-VLU-01, Section 5.1.4, the clearance section tag is not to be removed and the valve is not to be manipulated unless the specific conditions for manipulation of the valve are satisfied.

Distractor: a. Notify the CSO/SSS of discrepancy is **wrong** - Per Section 5.1.2 of N1-VLU-01, valves that are out of expected position with a clearance section tag installed are NOT discrepancies.

Distractor: c. Remove the Clearance Section Tag is **wrong** - Per GAP-OPS-02 Clearance Section tags are removed using the guidance provided on the Clearance Section Tag Removal Sheet. The presence of the tag is also to be noted on the valve lineup sheet under the INITIALS/DATE column.

Distractor: d. Update the Clearance Section Tag Removal Sheet is **wrong** - Again, N1-VLU-01 speaks to this in 5.1.4 The valve is absolutely NOT to be repositioned per GAP-OPS-02 until the Clearance Section Holder or their supervisor releases the clearance section. This will be indicated in the Clearance System and a separate lineup performed to restore the system called a RESTORATION LINEUP.

References Provided: NONE

Question 68 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 1

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- G2.1.29 3.4/3.3 Knowledge of how to conduct and verify valve lineups

Question Source
- Bank

Question Setting
- C1 (License class closed reference)

69 **SYSID: 21069** **Points: 1.00**

Given the following data from N1-ST-Q8B, Liquid Poison Pump 12 And Check Valve Operability Test:

8.27.2 WHEN a minimum of 5 minutes has elapsed, perform the following:

a. Simultaneously stop stopwatch AND flow Meter Totalizer. ___PB___

b. Record the following:

- Data Collection Time ___6___ mins (> 5 mins)
- Total Flow ___186___ gals ___EB___ IST

8.27.3 Calculate flow rate as follows:

Total gals __________ (Step 8.27.2.b) ÷ Total Mins __________ (Step 8.27.2.b) =

[TS] LIQUID POISON PUMP 12 flow rate ________gpm ________
(IST) (32 to 37 gpm)
[≥ 30 gpm]

____
Ind. Verif.

Which one of the following determinations can be made for LP 12 flow rate?

A. Flow rate is above the technical specification high limit.

B. Flow rate is below the technical specification low limit.

C. Flow rate is acceptable for the conditions observed.

D. Flow rate is outside the normal inservice test program range.

Answer: D

Associated objective(s):

Development Area (FIO)

Question 69 Details

Question Type:	Multiple Choice
Topic:	NRC RO 69
System ID:	21069
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-211-1-01, EO-1.7

Answer: d. 186 gallons/6 minutes = 31 gpm flow rate. This is outside the IST range indicated in parentheses ( ) and above the TS limit indicated with brackets [ ].
4.2 The letters IST in parenthesis (IST) will denote steps required for compliance with ASME Section XI, as delineated in the NMP1 Pump and Valve Inservice Testing Program Plan. If specific Acceptance Criteria is required, the acceptance value will be given in parenthesis, i.e., (30 sec).
4.3 The letters TS in brackets (TS) denotes steps required for compliance with Plant Technical Specifications. If specific Acceptance Criteria is required, then the acceptable value will be given in brackets, i.e. ($\geq$ 30 gpm)

Distractor: a. The TS limits is indicated in brackets [≥30 gpm] and is satisfactory with the flow rate at 31 gpm. The IST range is in parenthesis (32 to 37 gpm).

Distractor: b. The TS limits is indicated in brackets [≥30 gpm] and is satisfactory with the flow rate at 31 gpm. The IST range is in parenthesis (32 to 37 gpm).

Distractor: c. The IST range is in parenthesis (32 to 37 gpm) and the flow rate of 31 gpm is outside of this range.

References Provided: NONE

Question 69 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 3

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- G2.2.12 3/3.4 Knowledge of surveillance procedures

Question Source

- New

PROC

- N1-ST-Q8B, Rev. NA

Question Setting

- C1 (License class closed reference)

70 **SYSID: 21070** **Points: 1.00**

A clearance section tag for a system vent valve must be relocated so the valve can be removed from the system piping.

Which one of the following describes who must authorize the relocation before it occurs?

Tag relocation is required to be authorized by the .....

A. on-shift SSS alone.

B. CSO and CRS together.

C. CSO or CRS independently.

D. CSO or delegated licensed RO.

Answer: D

Associated objective(s):

Development Area (FIO)

Question 70 Details

Question Type:	Multiple Choice
Topic:	NRC RO 70
System ID:	21070
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	This question was derived from bank SYSID 17214.

Answer: d. CSO or delegated RO is **correct** - Tag relocation is under the purview of the Controller, defined in section 4.15 of GAP-OPS-02 as the CSO who performs the administrative function of issuing Clearance Sections as the agent for the SSS. The CSO may delegate this authority to another licensed RO per section 2.10.

Distractor: a,b&c Any option mentioning the SSS or CRS is **wrong** - the SSS responsibility as defined in GAP-OPS-02 is to authorize the isolation of equipment, ensure compliance with Tech Specs and verify adequacy of isolation boundaries.

References Provided: NONE

Question 70 Cross References (table item links)

10CFR55

- 41(b)(10)

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- G2.2.13 3.6/3.8 Knowledge of tagging and clearance procedures

Question Source

- Bank

PROC

- GAP-OPS-02 Rev. NA

Question Setting

- C1 (License class closed reference)

71 **SYSID: 21071** **Points: 1.00**

Given the following during a refueling outage:

- Phase 1 core shuffle is just completed
- In vessel work includes withdrawing a control rod to position 48 and uncoupling it to support change out of the control rod blade

Per N1-FHP-27C, Core Shuffle, which one of the following is an administrative requirement to be satisfied PRIOR TO withdrawing this control rod?

A. Jumpers installed to bypass all refueling interlocks.

B. Cell has no fuel and a double blade guide installed.

C. Caution tag posted on the rod motion control switch.

D. Clearance hung on this control rods hydraulic control unit.

Answer: B

Associated objective(s):

Development Area (FIO)

Question 71 Details

Question Type:	Multiple Choice
Topic:	NRC RO 71
System ID:	21071
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-234-1-01, EO-1.7

Answer: b. Per N1-FHP-27C; 1.1.5: Attachment 2 is performed to withdraw control rods in core cells that have been offloaded. This check provides a double verification that the **cell is empty** prior to rod withdrawal (i.e. Compliance with T.S.4.5.3), verifies defeat of the Refuel One Rod Permit Rod Block, and tags out the associated HCU per GAP-OPS-02.
Per N1-FHP-27C; Attachment 2: (1) **Cell verified empty**, (2) All other control rods in core cells containing one or more fuel assemblies are fully inserted, (3) All fuel loading operations have been suspended, (4) **Verify Double Blade guide is installed to support control rod during control rod withdraw**.

Distractor: a. Jumpers are installed to bypass the one rod permissive after the rod is withdrawn to position 48 and uncoupled. All refueling interlocks are not bypassed.

Distractor: c. Caution tag is posted on the rod motion control switch during fuel movement, not during control rod maintenance work.

Distractor: d. Clearance is hung on HCU 101 and 102 after control rod is withdrawn to position 48 and uncoupled.

References Provided: NONE

Question 71 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 2

Difficulty Level
- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- G2.2.26 2.5/3.7 Knowledge of refueling administrative requirements

Question Source
- New

PROC
- N1-FHP-27C Rev. NA

Question Setting
- C1 (License class closed reference)

72 **SYSID: 21072** **Points: 1.00**

The plant is at 35% power, with the following:

- A normal shutdown is in progress in preparation for an outage
- Chemistry reports no indications of airborne contamination in the containment
- The CRS has directed that containment de-inerting be commenced

Which one of the following identifies the vent path used when purging the drywell?

The purge will be from the drywell vent and purge valves, 201-31 and 201-32 to the ...

A. Main Stack via fan 201-35

B. RBEVS via BV 201-10 and 202-37(38)

C. Main Condenser via BV 201-11

D. RBVS via BV 201-18 and 202-36

Answer: A

Associated objective(s):

Development Area (FIO)

Question 72 Details

Question Type:	Multiple Choice
Topic:	NRC RO 72
System ID:	21072
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-223-1-04, EO-1.7

This questions was derived from bank SYSID 17687 EO-1.7.b

Answer: a. "stack..." is correct. The provided plant conditions indicate that the reactor is above 212°F with radioactivity below the limits requiring use of the RBEVS. Per N1-OP-9, Section G.1.0 and C-18014-C Sheet 1, the correct vent path to use is through the vent and purge fan to the stack.

Distractor: b. "RBEVS,,," is incorrect as this is not required by the conditions presented.

Distractor: c. "Main Condenser.." is incorrect as this vent path is not permitted under any normal operating procedure and may only be directed by the ED (see EOP bases and SDBD-202).

Distractor: d. "RBVS..." is incorrect as this does not provide adequate flow rate to provide suitable sweeping of nitrogen from the containment during the deinerting process.

References Provided: NONE

Question 72 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 1

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

DRW
- C-18013-C Rev. NA
- C-18014-C Rev. NA

NUREG 1123 KA Catalog Rev. 2
- G2.3.9 2.5/3.4 Knowledge of the process for performing a containment purge

Question Source
- Bank

PROC
- N1-OP-9 Rev. NA

Question Setting
- C1 (License class closed reference)

73 **SYSID: 21073** **Points: 1.00**

Given the following conditions:

- Your year-to-date exposure is 1800 mRem Total Effective Dose Equivalent (TEDE)
- You will be entering a Locked High Radiation Area
- The work will take forty-five (45) minutes to complete
- Work area radiation level is 1.6 Rem/hour
- The transient time EACH WAY is one (1) minute
- Transit path radiation level is 1.2 Rem/hour

Per GAP-RPP-07, Internal and External Dosimetry Program, which one of the following is the HIGHEST LEVEL OF AUTHORITY required to approve the dose extension to perform the work?

NOTE: Approvals are listed from lowest level to highest level of authority.

A. Station Shift Supervisor.

B. Rad Protection Manager

C. Plant General Manager.

D. Site Vice President.

Answer: D

Associated objective(s):

Development Area (FIO)

Question 73 Details

Question Type:	Multiple Choice
Topic:	NRC RO 73
System ID:	21073
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O3-OPS-006-343-3-40, EO-3.8

Answer: d. [(1600 mRem/hr) x (0.75 hr)] + [(1200 mrem/60 min) x (2 min)] = 1240 mRem
1240 mRem (projected exp.) + 1800 mRem (current exp.) = 3040 mRem
This is above 3000 mRem which requires site vice president approval.
Per GAP-RPP-07, 3.2.5: The following dose control levels have been established to prevent personnel from exceeding the administrative dose limits, and ensure equitable distribution of dose amongst workers who perform similar jobs. Appropriate station supervision and radiological protection management approval should be obtained to increase a worker's dose control level.
(1) 2,000 mrem per year--Radiological protection and line supervision approval and documented current year dose history are required to have the level increased.
(2) 2,500 mrem per year--The above approvals plus Radiation Protection Manager and Plant General Manager approval are required to have the level increased.
(3) 3,000 mrem per year--The above approvals plus notification of Site Vice President are required to have the level increased.

Distractor: a. Line supervisor is required for exposure from 2000-2499 mrem.

Distractor: b. Line supervisor is required for exposure from 2000-2499 mrem.

Distractor: c. Plant Manager is required for exposure from 2500-2999 mrem.

References Provided: NONE

Question 73 Cross References (table item links)

10CFR55
- 41(b)(12)

Cognitive Level
- 3

Difficulty Level
- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- G2.3.10 2.9/3.3 Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure

Question Source
- New

PROC
- GAP-RPP-07 Rev. NA

Question Setting
- C1 (License class closed reference)

74 **SYSID: 21074** **Points: 1.00**

Many control rods failed to insert when the reactor was scrammed.

Assuming the procedure steps will be successful in control rod movement, which one of the following EOP-3.1, Alternate Control Rod Insertion, methods will insert these control rods in the SHORTEST PERIOD of time?

A. Scramming control rods electrically.

B. Scramming control rods by venting the scram air header

C. Driving control rods by venting the over piston volume.

D. Driving control rods using the reactor manual control system.

Answer: A

Associated objective(s):

Development Area (FIO)

Question 74 Details

Question Type:	Multiple Choice
Topic:	NRC RO 74
System ID:	21074
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-344-1-11, EO-1.2

Answer: a. Performed if any group solenoid or Backup Scram light is lit; consists of pulling fuses in the aux control room to insert all control rods simultaneously. Remove five (5) fuses in cabinet 1S-53 AND remove five (5) fuses in cabinet 1S-55 then all control rods insert.

Distractor: b. Consists of the following:
1. Unlock and Close 113-3091, (Scram Air Header Supply valve, located in the Reactor Building, NW stairwell, between Elev. 237 and 261).
2. Remove the Vent Pipe Cap from 113-230, SCRAM AIR HEADER EMERGENCY VENT VALVE, RB El 237 (located at northwest corner of HCU Bank).
3. Unlock AND Open 113-230, SCRAM AIR HEADER EMERGENCY VENT VALVE
The fuses in the aux control room can be pulled before obtaining the tools to perform this method.

Distractor: c. The over piston area of the control rods are vented one at a time. This is very time consuming compared to pulling fuses to insert all control rods.

Distractor: d. The control rods are driven one at a time. This is very time consuming compared to pulling fuses to insert all control rods.

References Provided: NONE

Question 74 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 2

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- G2.4.7 3.1/3.8 Knowledge of event based EOP mitigation strategies

Question Source

- New

PROC

- N1-EOP-3.1 Rev. NA

Question Setting

- C1 (License class closed reference)

75 **SYSID: 21075** **Points: 1.00**

A loss of Service Water has occurred, with the following:

- H1-4-2, R BUILDING SW PRESS/SERV W PUMP HDR PRESS LOW is in alarm
- N1-SOP-18.1, Service Water Failure / Low Intake Level is entered
- With both service water pumps running service water header pressure is 20 psig and steady

Per N1-SOP-18.1, which one of the following actions will be directed NEXT by the ASSS?

A. Start Emergency Service Water pumps.

B. Place the reactor mode switch to shutdown.

C. Supply fire water to closed loop cooling heat exchangers.

D. Lower reactor power to control closed loop cooling temperatures

Answer: A

Associated objective(s):

Development Area (FIO)

Question 75 Details

Question Type:	Multiple Choice
Topic:	NRC RO 75
System ID:	21075
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-276-1-01, EO-1.7

Answer: a. SW header pressure cannot be restored requiring start of ESW and then a reactor scram

IF	THEN
Neither Service Water pump can be started, OR Service Water Hdr pressure can NOT be restored, AND As directed by SRO,	1. Start Emergency Service Water pumps. 2. Scram the Reactor. 3. Perform N1-SOP-1 concurrently. 4. Initiate Emergency Condensers. 5. Close MSIVs. 6. Trip all Rx Recirculation Pumps.

Distractor: b. ESW pumps are started then the reactor scram is inserted.

Distractor: c. This is required if the ESW pumps cannot be started. This condition cannot be determined at this time.

Distractor: d. A reactor scram is required, not just a power reduction. This action is appropriate is service water header pressure returned to normal but it is below the acceptable operating range.

References Provided: NONE

Question 75 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 2

Difficulty Level
- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- ~~G2.2.12~~ 3/3.4 ~~Knowledge of surveillance procedures~~

typo 2.4.25 Loss of Cooling Water Procedure

Question Source
- New

PROC
- N1-SOP-18.1, Rev. NA

Question Setting
- C1 (License class closed reference)

EXAMINATION ANSWER KEY

1P

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

1 **SYSID: 21076** **Points: 1.00**

The plant is at 100% power. The Static Battery Charger (SBC) alignment will be changed as follows:

- SBC 161A will be removed from service for 4 hours for maintenance.
- SBC 161A will be transferred to SBC 161B per N1-OP-47A, Section F.2.0, Shifting from SBC 161A to SBC 161B.

Which one of the following is the correct application of TS 3.6.3 for the above conditions including the justification for the action?

A. It is not necessary to enter the actions of TS 3.6.3 because MG 167 is available as an alternate battery charger before and after aligning SBC 161B for service.

B. Enter TS 3.6.3 because this battery system is inoperable for the time period from when SBC 161A is disconnected until SBC 161B is connected for service.

C. Enter TS 3.6.3 for a loss of battery system and continue in the TS action once the transfer is complete. Restore SBC 161A to operable within 24 hours to avoid a plant shutdown.

D. It is not necessary to enter the actions of TS 3.6.3 because SBC 161B is connected to this battery system before disconnecting SBC 161A maintaining this battery system operable.

Answer: B

Associated objective(s):

Development Area (FIO)

Question 1 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 1
System ID:	21076
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00

Comment:

Objective: O1-OPS-001-263-1-01, EO-1.8.a

Answer: b. The transfer from SBC 161A to SBC 161B requires disconnecting SBC 161A from the battery system before connecting SBC 161B to the battery system. With no battery charger in service (both disconnected from the battery system) the battery system is considered inoperable and TS 3.6.3 applies. Once SBC 161B is in service, TS 3.6.3 LCO can be exited.

Distracter: a. Although MG 167 can be used as a battery charger, it is not safety related and cannot be used to avoid entry into TS 3.6.3 LCO.

Distracter: c. This is the correct action if both SBC 161A/B or 171A/B are removed from service: Removal of both Static Battery Chargers for a battery system are treated as loss of a battery system. A battery charger must be returned to service within 24 hours per T.S. 3.6.3.h or take the action required by T.S. 3.1.5 which requires reactor coolant pressure be reduced to 110 psig or less and reactor coolant temperature be reduced to saturation temperature or less within 10 hours. Although MG 167 can be used as a battery charger, it is not safety related and cannot be used to exit the LCO.

Distracter: d. The transfer from SBC 161A to SBC 161B requires disconnecting SBC 161A from the battery system before connecting SBC 161B to the battery system. Momentary TS 3.6.3 entry is required

References Provided: None

N1-OP-47A; F.3.0, D.9.0

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 1 Cross References (table item links)

10CFR55

- 43(b)(2)

Cognitive Level

- 2

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 295004 AA2.03 2.8/2.9 Battery voltage
AA2.03 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Battery voltage.

Question Source

- New

PROC

- N1-OP-47A Rev. NA

TECHSPEC

- 3.6.3 Rev. NA

Question Setting

- C1 (License class closed reference)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

77

2 **SYSID: 21077** **Points: 1.00**

The plant is in a refueling outage ready to start the first fuel shuffle. Before starting fuel movement, Shutdown Cooling (SDC) is lost.

w

~~Per N1-SOP-20, Loss of SFP / Reactor Cavity Level / Decay Heat Removal~~, which one of the following is required if actual temperature measurements indicate reactor water temperature is projected to exceed 140°F?

A. Perform a "time to boil" estimation.

B. ~~Perform a "shutdown margin" demonstration~~. Perform a feed & bleed to spent fuel pool.

C. Establish conditions for entering Hot Shutdown.

D. Establish conditions for returning to Cold Shutdown.

Answer: A

Associated objective(s):

Development Area (FIO)

Question 2 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 2
System ID:	21077
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-205-1-01, EO-1.8.a

Answer:	a. Per N1-SOP-20, with safety assessment level for DHR <N+1 and temperature of the reactor projected to exceed 140°F, perform a time to boil estimation per N1-ODP-OPS-0108.
Distracter:	b. This is not a requirement for the conditions specified. SDM demonstration is performed following loading of the reactor core. For these conditions, SDM is plausible because its definition includes moderator temperature at 68°F. This is a large change from the analysis temperature but in a conservative direction.
Distracter:	c. Hot shutdown equates to reactor water temperature > 212°F. This is not a concern at this time and is not the appropriate action per N1-SOP-20.
Distracter:	d. The main difference between refueling and cold shutdown is the reactor mode switch can be in either shutdown or refuel for cold shutdown and must be in refuel for the Refueling Condition. For the cold shutdown condition the reactor mode switch can be placed to startup to perform the shutdown margin demonstration. There is no benefit to returning to cold shutdown and this is not the appropriate action per N1-SOP-20

References Provided: None

Question 2 Cross References (table item links)

10CFR55

- 43(b)(~~6~~) 5 X

Cognitive Level

- 1

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 295021 AA2.04 3.6/3.5 Reactor water temperature
AA2.04 Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor water temperature.

Question Source

- New

PROC

- N1-SOP-20 Rev. NA

Question Setting

- C1 (License class closed reference)

EXAMINATION ANSWER KEY

18

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

3 **SYSID: 21078** **Points: 1.00**

Assume the design REFUELING ACCIDENT as described in UFSAR Chapter XV occurs on Unit One (1).

Which one of the following effects of ~~this~~ the above events results in the event being reportable per 10CFR50.72?

A. Lowering fuel pool level cannot be restored above 338 feet.

B. Rising Source Range Monitor count rate indicates criticality.

C. Lowering reactor cavity level causes the uncovery of irradiated fuel.

D. Rising refuel floor radiation levels cause upscale instrument readings.

Answer: D

Associated objective(s):

Development Area (FIO)

Question 3 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 3
System ID:	21078
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-234-1-01, EO-1.7.d

Answer: d. The REFUELING ACCIDENT constitutes a dropped irradiated fuel assembly from the highest elevation that it can be dropped from using refueling equipment (~30feet) resulting in the perforation of approximately 105 fuel rods. Rising refuel floor radiation levels result from the fuel damage. EAL Matrix classification 1.4.1,1.4.2.

Distractor: a. Lowering fuel pool level is not the accident sequence for the REFUELING ACCIDENT. However it is addressed on the EAL Matrix: 1.5.1.

Distractor: b. Inadvertent criticality is not the accident sequence for the REFUELING ACCIDENT. However, it is a concern during refueling and likely that it could be considered to be a refueling accident.

Distractor: c. Lowering reactor cavity and uncovery of irradiated fuel is not the accident sequence for the REFUELING ACCIDENT. However it is addressed on the EAL Matrix: 1.5.2

References Provided: EAL MATRIX

Question 3 Cross References (table item links)

10CFR55
- 43(b)(1)
- 43(b)(4)

Cognitive Level
- 2

Difficulty Level
- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- G2.4.30 2.2/3.6 Knowledge of which events related to system operations/status should be reported to outside agencies
- 295023 Refueling Accidents

Question Source
- New

Question Setting
- C1 (License class closed reference)

19

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

4 **SYSID: 21079** **Points: 1.00**

The plant was at 100% power when a manual reactor scram was inserted but many control rods failed to insert. Conditions are:

- Bypass valves are controlling reactor pressure
- LP is initiated and injecting
- All RPV injection is prevented, except for CRD and LP
- RPV injection is reestablished with a band of -54 to -9~~7~~0 inches inl. fuel zone
- Reactor Power is currently 10%
- ~~Reactor pressure is 920~~ psig

Subsequently, a loss of vacuum occurs. ERVs are open and torus water temperature is 114°F.

Per N1-EOP-3, FAILURE TO SCRAM, which one of the following is the required EOP level control step (indicate the step number) to be performed at this time?

A. Go to point ⑧ and perform the actions of step L-7.

B. Continue at point ⑨ and perform the actions of step L-9.

C. Re-enter at point ❻ and continue the current actions of step L-6.

D. Re-enter at point ❼ and continue the current actions of step L-6.

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 4 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 4
System ID:	21079
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-344-1-03, EO-1.2

Answer:	a. ERV opening completes the requirements to enter ⑧. Terminate and prevent injection until ERVs are closed or reactor power is below 6% or RPV level lowers to TAF.
Distractor:	b. Based on the conditions provided, this is the step being performed before Torus water temperature exceeds 110°F.
Distractor:	c. Overrides are continually evaluated, they are not re-entered, must go to⑧.
Distractor:	d. Overrides are continually evaluated, they are not re-entered, must go to ⑧.

References Provided: EOP-3

Question 4 Cross References (table item links)

10CFR55

- ~~43(b)(5)~~ 41.b.10

Cognitive Level

- 3

Difficulty Level

- Level 4: Highest order knowledge item requiring use of problem solving skills, judgement, and maximum task complexity as well as applying procedures to determine correct answer.

NUREG 1123 KA Catalog Rev. 2

- G2.4.6 3.1/4 Knowledge symptom based EOP mitigation strategies

- 295025 High Reactor Pressure

Question Source

- Bank

PROC

- N1-EOP-3 Rev. NA
- N1-ODP-PRO-0305 Rev. NA

Question Setting

- C1 (License class closed reference)

8D

5 **SYSID: 21080** **Points: 1.00**

A steam leak in the drywell has occurred. Primary containment parameters for the past four (4) minutes are as follows:

08:01:	Drywell Pressure	6.0 psig
	Torus Pressure	5.0 psig
	Drywell Temperature	225°F
08:02:	Drywell Pressure	8.0 psig
	Torus Pressure	7.0 ~~psig~~
	Drywell Temperature	250°F 300
08:03:	Drywell Pressure	11.0 psig
	Torus Pressure	10.0 psig
	Drywell Temperature	276°F
08:04:	Drywell Pressure	14.0 psig
	Torus Pressure	13.0 psig
	Drywell Temperature	302°F

Which one of the following is the EARLIEST TIME at which containment spray can be initiated?

A. 08:01

B. 08:02

C. 08:03

D. 08:04

Answer: B

Associated objective(s):

Development Area (FIO)

Question 5 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 5
System ID:	21080
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00

Comment:

Objective: O1-OPS-006-344-1-04, EO-1.5

Answer: b. At 250°F, drywell pressure must be above @7.3 psig to spray. This requirement is met for these conditions. Torus pressure is not used in the spray determination. If use torus pressure would determine still in the "no spray" region and select answer "c" or "d" rather than this answer. Although it could be inferred that although in the safe region of the drywell spray initiation limit, that spray cannot be initiated until torus pressure reaches 13 psig which is a condition requiring containment spray, this is incorrect because required to spray prior to reaching 300°F.

Distractor: a. At 225°F, drywell pressure must be above @6.8 psig to spray. Torus pressure is not used in the spray determination but could be inadvertently used but should determine still in "no spray" region

Distractor: c. At 276°F, drywell pressure must be above @7.8 psig to spray. This requirement is not met for drywell pressure but if torus pressure is inadvertently used could determine in "okay to spray" region.

Distractor: d. At 302°F, drywell pressure must be above @8.4 psig to spray. This requirement is met for these conditions however the "okay to spray" region was entered earlier. Torus pressure is not used in the spray determination. It could be inferred that although in the safe region of the drywell spray initiation limit, that spray cannot be initiated until torus pressure reaches 13 psig which is a condition requiring containment spray,

References Provided: EOP-4

Question 5 Cross References (table item links)

10CFR55
- 43(b)(5) ✓

Cognitive Level
- 2

Difficulty Level
- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- 295028 EA2.01 4.0*/4.1* Drywell temperature
Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Drywell temperature.

Question Source
- New

PROC
- N1-EOP-04 Rev. NA

Question Setting
- C1 (License class closed reference)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

6 **SYSID: 21081** **Points: 1.00**

Which one of the following primary containment challenges requires entry into EOP-8, RPV Blowdown?

A. Torus water temperature rises and is within 0.5°F of the boron injection initiation temperature.

B. Hydrogen concentration and oxygen concentration each have reached the flammability limit.

C. Torus water level lowers and is approaching the level at which openings in the ERV discharge devices become uncovered.

D. Drywell temperature rises to within 10°F of the environmental qualification temperature for ADS during a reactor coolant leak.

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 6 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 6
System ID:	21081
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-344-1-04, EO-1.2

Answer:	c. RPV blowdown is required before torus level reaches 8 feet, the level at which the openings in the ERV discharge devices become uncovered to avoid direct pressurization of the torus air space when ERVs are operated.
Distractor:	a. BIIT is 110°F. This temperature only requires a scram and boron injection. If the HCTL is exceeded, then a RPV Blowdown is required.
Distractor:	b. Actions such as venting are taken when concentrations reach the flammability limit. RPV blowdown is not taken until the deflagration limit is exceeded.
Distractor:	d. This equates to 291°F since the EQ temperature for ADS is 301°F. RPV blowdown on drywell temperature is only required if drywell temperature cannot be restored and maintained below 300°F. Reactor pressure remains high during a coolant leak and therefore the unsafe region of the saturation curve for level instruments is not challenged until after RPV blowdown.

References Provided: EOP-4

Question 6 Cross References (table item links)

10CFR55

- 43(b)(5) ✓

Cognitive Level

- 1

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- G2.4.6 3.1/4 Knowledge symptom based EOP mitigation strategies

- 295030 Low Suppression Pool Wtr Lvl

Question Source

- New

PROC

- N1-EOP-4 Rev. NA
- N1-ODP-PRO-0305 Rev. NA

Question Setting

- C1 (License class closed reference)

82

7 **SYSID: 21082** **Points: 1.00**

During an ATWS, which one of the following is the CORRECTED FUEL ZONE water level that equates to the Minimum Steam Cooling RPV Water Level for maintenance of adequate core cooling if reactor power is 11%?

A. - 94 inches

B. - 99 inches

C. - 122 inches

D. - 124 inches

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 7 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 7
System ID:	21082
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00

Comment:

Objective: O1-OPS-006-344-1-03, EO-1.5

N1-ODP-PRO-0305, EOP Figure X

Answer: c. **[-122 inches]**
The Minimum Steam Cooling RPV Water Level is the lowest RPV water level at which the covered part of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1500° F. The level is used to preclude fuel damage when RPV water level is below the top of the active fuel. The calculated value for NMP1 is -109 in.
The actual RPV water level value for MSCWL is -109 inches. The corrected value is -122 inches. Correction factor for 10% power is -13 inches (with reactor power between 10% and 12% would use conservative value of 10% which provides a higher reactor water level).

-14" fn 11% pwr

Is this more conservative

Distractor: a. **[-94 inches]** Corrected value for TAF not the MSCRWL if the incorrect data is used. Use correction factor of -10 inches rather than 10% reactor power.

Distractor: b. **[-99 inches]** Corrected value for MSCRWL if the incorrect data is used. Use correction factor for 10% reactor power but applied to TAF not the MSCWL.

Distractor: d. **[-124 inches]** Correction factor for 10% power is -13 inches (with reactor power between 10% and 12% would use conservative value of 10% which provides a higher reactor water level not 12%).

References Provided: EOP-3

Question 7 Cross References (table item links)

10CFR55

- 43(b)(5) ✓

Cognitive Level

- 2

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 295031 EA2.04 4.6*/4.8* Adequate core cooling
Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL

Question Source

- New

PROC

- N1-EOP-3 Rev. NA
- N1-ODP-PRO-0305 Rev. NA

Question Setting

- C1 (License class closed reference)

8 **SYSID: 21083** **Points: 1.00**

The plant is operating at 100% power when the Fire Detection System senses a fire in Hazard C-2123, Power Board Room 102. The following Alarm Detection Zones are received at the Main Fire Control Panel:

- DX-2123A
- DX-2123B

An AO reports:

- Local horn and light actuate
- Thirty (30) seconds later CO2 discharges into the area

If ~~plant conditions are~~ stable, which one of the following is required at this time in response to the above conditions?

A. Direct the CSO to implement EPIP-EPP-28, Attachment 1, CSO Fire Fighting Checklist.

B. Declare an unusual event per EAL 8.2.1 and enter EPIP-EPP-08, Activation and Direction of the Emergency Plans.

C. Dispatch single FB member to check fire alarm before Delegate ~~the fire brigade~~ response to be coordinated through the Operations Support ~~Center (OSC)~~ director. dispatching Brigade

D. Direct the CSO to scram the reactor and direct the actions of the applicable abnormal and emergency operating procedures.
Direct CSO to scram the reactor because ability to establish safe s/d condition is lost

Answer: A

Would be appropriate if not auto emulation

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 8 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 8
System ID:	21083
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Comment:

Objective: O1-OPS-001-286-1-05, EO-1.7.d

EPIP-EPP-28, 3.2.1.a, 4.1

Answer: a. When credible evidence exists of a fire condition within the protected area, then per EPIP-EPP-28, direct the CSO to implement the CSO fire fighting checklist.
The definition of CONFIRMED FIRE is a condition in which credible evidence exists that a fire is actually occurring. A fire may be considered as confirmed given any of the following: fire alarm/annunciator AND suppression system activation accompanied by actual flow or discharge, or Fire Brigade/Leader report, or SSS judgement.

Distractor: b. An unusual event is not declared until confirmed fire not extinguished within 15 minutes of control room notification. There is no evidence that the fire is not extinguished and 15 minutes have not expired.

Distractor: c. This is a correct action if the OSC is operational. The OSC is not operational under these conditions but could be summoned to activate if the fire is not extinguished within 15 minutes.

Distractor: d. Per N1-SOP-9, if either of the following conditions exist due to the fire: spurious valve operation or loss of equipment control or fire not under control within 15 minutes or fire endangers safe shutdown capability, the initiate a reactor scram and enter SOP-1 and EOP-2. None of these conditions are met or can be determined conclusively at this time.

References Provided: EAL Matrix

Question 8 Cross References (table item links)

10CFR55
- 43(b)(5)

Cognitive Level
- 1

Difficulty Level
- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- G2.1.2 3/4 Knowledge of operator responsibilities during all modes of plant operation
- 600000 Plant Fire On Site

Question Source
- New

PROC
- EPIP-EPP-28 Rev. NA

Question Setting
- C1 (License class closed reference)

9 **SYSID: 21084** **Points: 1.00**

A reactor startup is in progress. N1-ST-C2, Solenoid-Actuated Pressure Relief Valves Operability And Flow Verification Test will be performed. Torus average temperature prior to the start of the test is 82°F.

Which one of the following is the threshold torus average temperature that if exceeded during the test requires declaring the Tech Spec LCO Statement NOT met and entering a TS Action Statement?

A. 85°F.

B. 90°F.

C. 92°F.

D. 95°F.

Answer: D

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 9 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 9
System ID:	21084
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-226-1-01, EO-1.11

Answer:	d. Per TS 3.3.2.d, during testing of relief valves that add heat to the torus the operating limit of 85°F is raised 10°F to 95°F.
Distractor:	a. This is the normal operating limit without testing of relief valves that add heat to the torus. The operating limit of 85°F is raised 10°F to 95°F during this testing.
Distractor:	b. Could add 5°F to the operating limit and determine the limit to be 90°F. The 5°F value could be confused with the requirement to monitor torus water temperature every 5 minutes during the test.
Distractor:	c. The operating limit is raised 10°F to 95°F during this testing. Could add 10°F to the starting temperature and determine the limit to be 92°F.

References Provided: None

Question 9 Cross References (table item links)

10CFR55
- 43(b)(2)

Cognitive Level
- 1

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- G2.2.22 3.4/4.1 Knowledge of limiting conditions for operations and safety limits
- 295013 High Suppression Pool Temp

Question Source
- New

TECHSPEC
- 3.3.2.D Rev.

Question Setting
- C1 (License class closed reference)

10 **SYSID: 21085** **Points: 1.00**

An ATWS is in progress. EOP-3, Failure To Scram, is being executed including actions to insert control rods. After inserting a manual reactor scram all control rods insert except one control rod which is at position 24 and will NOT move.

Which one of the following is required to exit from EOP-3 if any?

A. The reactor engineer must determine the reactor will remain shutdown without boron.

B. No additional action is needed since the reactor will remain shutdown without boron.

C. Continue boron injection until the equivalent of hot shutdown boron is injected into the reactor.

D. Continue boron injection until the entire contents of the Liquid Poison Tank are injected into the reactor.

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 10 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 10
System ID:	21085
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-344-1-03, EO-1.3 & 1.4

Answer: b. Per N1-ODP-PRO-0305; if one or more rods are withdrawn past position 04, other criteria may be used to determine the reactor will remain shutdown. These include TS reactivity margin criteria and rod pattern analysis performed by the reactor engineer. Per Tech Specs, the shutdown margin is met when all control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn (position 48). This requirement is met and EOP-3 can be exited without further action.

Distractor: a. The reactor engineer could be asked to perform a rod pattern analysis if more than one control rod was withdrawn beyond 04 to make the determination that the reactor is shutdown and will remain shutdown without boron however this is not required since only one rod is not fully inserted (beyond 04).

Distractor: c. This permits raising RPV water level to +53 to +95 inches but does not permit exiting EOP-3. This exit condition is already satisfied.

Distractor: d. This permits exiting EOP-3 power leg and entering N1-SOP-1 but is not required since the exit condition is already satisfied.

References Provided: None

Question 10 Cross References (table item links)

10CFR55
- 43(b)(2)
- 43(b)(5)
- 43(b)(6)

Cognitive Level
- 1

Difficulty Level
- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- G2.1.23 3.9/4 Ability to perform specific system and integrated plant procedures during different modes of plant operation
- 295015 Incomplete Scram

Question Source
- New

PROC
- N1-ODP-PRO-0305 Rev. NA

TECHSPEC
- 1.32, Rev. NA

Question Setting
- C1 (License class closed reference)

11 **SYSID: 21086** **Points: 1.00**

The plant is at 100% power with fuel leaks identified:

15:00	K1-1-2, EMER COND VENT 11 RAD MONITOR, alarms
15:01	EMERG COND RMON 111 and 112 on J panel are both in alarm
15:02	CRS directs EC 11 be isolated but it CANNOT be isolated
15:05	EMERG COND RMON 111 and 112 are steady at 28 mrem/hr
15:06	Manual reactor scram inserted
15:07	EMERG COND RMON 111 and 112 are at 15 mrem/hr and lowering slowly

Which one of the following is the required action AT THIS TIME?

A. Cooldown at a rate below 100°F/hr as directed by EOP-2.

B. Cooldown at a rate above 100°F/hr as directed by EOP-2.

C. Perform a RPV Blowdown per EOP-8 as directed by EOP-5.

D. Perform a RPV Blowdown per EOP-8 as directed by EOP-6.

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 11 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 11
System ID:	21086
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Comment:

Objective: O1-OPS-001-207-1-01, EO-1.8.b

Answer: a. Reduce reactor pressure to lower the release. No conditions present at this time that warrant actions beyond a normal cooldown.

Distractor: b. Acceptable to exceed 100°F/hr if anticipating emergency depressurization. Conditions do not warrant exceeding the TS cooldown rate at this time. If the radiation monitor readings continued to rise, then it might be reasonable to determine that alert release rate would be exceeded and higher (site area emergency and general emergency) classifications may be challenged.
RPV blowdown is anticipated
As conditions which will require a blowdown are approached it is appropriate to rapidly reject as much heat energy as possible from the RPV to a heat sink other than the torus. Such action preserves the heat capacity of the torus for as long as possible, until a requirement for a blowdown actually exists.
"Anticipated" implies an expectation, based on an evaluation of plant conditions and extrapolation of parameter trends, that a blowdown requirement will soon be reached and cannot be averted by actions prescribed in the EPGs. Before this conclusion can be drawn, however, the effectiveness of the steps preceding the blowdown requirement must be evaluated. For example, if drywell temperature is increasing, a blowdown should not be "anticipated" in the Drywell Temperature branch of EOP-4 until the steps addressing operation of drywell cooling and drywell sprays have been performed.

Distractor: c. EOP-5 is likely entered based on "shine" causing ARM on west end of shield wall 340' elevation to exceed the EOP-5 entry set point. However, a primary system is not discharging into the reactor building. It is discharging outside the reactor building, therefore, cannot achieve blowdown requirement from EOP-5 based on the conditions presented.

Distractor: d. EOP-6 is not entered until the ALERT level is achieved within the e-plan. Both radiation monitors rise above the unusual event level and challenge the alert release level then lower when the reactor is scrammed. Additionally, because this is an unmonitored release, need field survey data before determining a blowdown required per EOP-6.

References Provided: EOP-2, EOP-5/6

Question 11 Cross References (table item links)

10CFR55

- 43(b)(4)
- 43(b)(5)

Cognitive Level

- 3

Difficulty Level

- Level 4: Highest order knowledge item requiring use of problem solving skills, judgement, and maximum task complexity as well as applying procedures to determine correct answer.

NUREG 1123 KA Catalog Rev. 2

- 295017 AA2.01 2.9*/4.2* Off-site release rate: Plant-Specific

AA2.01 Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE : Off-site release rate.

Question Source

- New

PROC

- N1-EOP-02 Rev. NA
- N1-EOP-05 Rev. NA
- N1-EOP-06 Rev. NA

Question Setting

- C1 (License class closed reference)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

12 **SYSID: 21087** **Points: 1.00**

The plant is at 80% power when a seismic event occurs. Subsequently, because of a torus water leak the following are observed:

- H2-2-1, R BLDG FL DR SUMPS 11-16 AREA WTR LVL LEVEL HIGH, in alarm
- Computer Pt. F188 NE RB CORNER RM WTR LVL HIGH in alarm

Which one of the following is the operability of the safety-related pumps in this area?

~~A.~~ Core Spray Pumps 121 and 122 are inoperable at this time.

~~B.~~ Core Spray Pumps 121 and 122 remain operable until level in this area rises an additional 2 feet.

C. Containment Spray Pumps 112 and 122 are inoperable at this time.

D. Containment Spray Pumps 112 and 122 remain operable until level in this area rises an additional 2 feet

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 12 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 12
System ID:	21087
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-344-1-05, EO-1.2

N1-ODP-PRO-03-5, element SC-05 through SC-08

Answer:	c. Containment Spray Pumps 112 and 122 are in the NE corner room and are the components affected by the water level in the room. The alarm is actuated at a water level of 5 feet in the room, which is the maximum safe value. The max safe value is defined to be the highest value at which equipment necessary for the safe shutdown of the plant will operate. Therefore the components in the area are inoperable.
Distractor:	a. Core Spray Pumps 121 and 122 are in the SE corner room and are not affected
Distractor:	b. The maximum safe value is already reached. Therefore the components in the area are inoperable. Core Spray Pumps 121 and 122 are in the SE corner room and are not affected.
Distractor:	d. The maximum safe value is already reached. **References Provided: None**

Question 12 Cross References (table item links)

10CFR55

- 43(b)(2)
- 43(b)(5)

Cognitive Level

- 2

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 295036 EA2.01 3/3.2 Operability of components within the affected area.
Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Operability of components within the affected area.

Question Source

- New

PROC

- N1-ODP-PRO-0305 Rev. NA
- N1-ARP-H2-2-1, Rev. NA

Question Setting

- C1 (License class closed reference)

13 **SYSID: 21088** **Points: 1.00**

The plant is at 100% power with the following inoperable equipment:

- EC 11 is not restored to operable status within the 7 days permitted by TS 3.1.3.b.
- At 12:00 you enter the action of TS 3.1.3.e which states "a normal orderly shutdown shall be initiated within one hour, and the reactor shall be in the cold shutdown condition within 10 hours."
- At 12:30, recirc flow has been reduced to 41×10^6 lbm/hr, cram rods have been inserted, and the crew has commenced inserting control rods per the Control Rod Sequence.

Which one of the following actions is required to achieve cold shutdown as required by TS 3.1.3.e without violating any procedure or TS limits?

A. Insert a manual scram no later than 15:00 and establish Shutdown Cooling permissives no later than 19:00.

B. Insert a manual scram no later than 15:30 and establish Shutdown Cooling permissives no later than 21:00.

C. Authorize cool down rates up to 100°F/hr to avoid a reactor scram and establish shutdown cooling permissives no later than 19:30.

D. Authorize cool down rates up to 100°F/hr to avoid a reactor scram and establish shutdown cooling permissives no later than 22:00

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 13 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 13
System ID:	21088
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-006-APS-1-01, EO-1.2 and O1-OPS-008-362-1-03, EO-1.11

N1-OP-43C, D.19

Answer: a. When required to shutdown and cool down to meet a 10-hour LCO requirement, it is necessary insert a manual scram approximately 3 hours into the LCO. It is also necessary to have SDC permissives met approximately 7 hours into the LCO to achieve cold shutdown in 10 hours.

Distractor: b. SDC at 21:00 only leaves 1 hour to reduce temperature about 140 degrees exceeding the cooldown rate. The scram is to occur within 3 hours of the TS3.1.3.e entry, not the 12:30 conditions stated

Distractor: c &d, cold shutdown cannot be achieved within 10 hours without performing the reactor scram to accelerate the shutdown. Without a manual reactor scram, limits would be violated to meet the 10-hour requirement. Raising the cooldown rate above the administrative limit up to the TS limit does not aid in obtaining cold shutdown within 10 hours of entering TS 3.1.3.e.

References Provided: None

Question 13 Cross References (table item links)

10CFR55

- 43(b)(5)
- 43(b)(6)

43(b)(2)

Cognitive Level

- 3

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- G2.2.22 3.4/4.1 Knowledge of limiting conditions for operations and safety limits
- 207000 Isolation (Emergency) Condenser

Question Source

- New

PROC

- N1-OP-43A Rev. NA

TECHSPEC

- 3.1.3 Rev. NA

Question Setting

- C1 (License class closed reference)

14 **SYSID: 21089** **Points: 1.00**

Unit One (1) is at 100% power. At 08:00 (12/1/2004), K1 2-1, LIQ POISON EXPLOSIVE VALVE 11-12 CONTINUITY, alarms. The following are verified for the Liquid Poison (LP) System:

- LP System 11 explosive valve light on Panel K is OFF.
- LP System 11 ammeter located at 1S-65 indicates 0 milliamps

- LP System 12 explosive valve light on Panel K is ON.
- LP System 12 ammeter located at 1S-65 indicates 0.95 milliamps.

Per Technical Specifications, which one of the following is a required action in response to the conditions above?

A. Initiate an orderly shutdown no later than 09:00 on 12/1/2004

B. Complete N1-ST-M1B, Liquid Poison Pump 12 Operability Test, no later than 08:00 on 12/2/2004.

C. Immediately verify the redundant component is operable and verify it remains operable once every 24 hours thereafter.

D. Immediately verify the liquid poison solution parameters are within the equivalency equation and TS Figure 3.1.2.b.

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 14 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 14
System ID:	21089
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00

Comment:

Objective: O1-OPS-001-211-1-01, EO-1.11

Answer: c. Per TS 3.1.2.b, if a redundant component becomes inoperable, specification 3.1.2.a shall be considered fulfilled, provided that the component is returned to an operable condition within 7 days and the additional surveillance required is performed.
Per the TS bases, the explosive valves are considered redundant components. The functional test and other surveillances on components, along with monitoring instrumentation, demonstrates system operability. The continuity check of the firing circuit on the explosive valves is one of the means to ensure system operability.
Per Technical Specification 4.1.2.c, SURVEILLANCE WITH INOPERABLE COMPONENTS, when a component becomes inoperable its redundant component shall be verified to be operable <u>immediately</u> and <u>daily</u> thereafter.
TS 3.1.2.b actions apply because the explosive valve is a redundant component. The TS requirement is to restore it to operable in 7 days and to immediately check the redundant component (the other explosive valve) immediately and then daily until the LCO statement is met.

Distractor: a. TS 3.1.2.b actions apply because the explosive valve is a redundant component. If not a redundant component, then TS 3.1.2.e actions would be entered.

Distractor: b. Monthly surveillance only tests the pump and not the explosive valve. The requirement is to test the redundant component. Additionally, the monthly surveillance results in LP System 12 being inoperable and a TS 3.0.1 entry which is not prudent; intentional entry into TS 3.0.1 is prohibited.

Distractor: d. The solution parameters are not required to be verified; only the redundant component. If both valves inop, then this would be required.

References Provided: TS 3.1.2 and TS 4.1.2
DO NOT provide the bases. DO NOT provide figure 3.1.2b

Question 14 Cross References (table item links)

10CFR55
- 43(b)(2)

Cognitive Level
- 2

Difficulty Level
- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- G2.2.25 2.5/3.7 Knowledge of bases in technical specifications for limiting conditions for operations and safety limits
- 211000 SLC

Question Source
- New

TECHSPEC
- 3.1.2 Rev. NA
- 3.1.2b Rev.
- 4.1.2.c, Rev. NA

Question Setting
- C1 (License class closed reference)

15 **SYSID: 21090** **Points: 1.00**

Phase II core shuffle in progress.

- The next step (step 226) requires movement of the fuel assembly at core location 28-19 to core location 24-39.
- Before latching step 226, SRM 11 and SRM 13 are declared inoperable.

Per the Tech Spec SRM operability requirements for refueling, which one of the following is the reason why step 226 CANNOT be performed?

A. The fuel assembly would be moved to a core quadrant with an inoperable SRM.

B. The fuel assembly would be removed from a core quadrant with an inoperable SRM.

C. An SRM must be operable in each of the four (4) core quadrants when core alterations are in progress.

D. An SRM must be operable in one (1) of the core quadrants adjacent to the locations of the core alterations.

Answer: D

Associated objective(s):

Development Area (FIO)

Too Leading for An SRO Applicant

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 15 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 15
System ID:	21090
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-234-1-01, EO-1.11, EO-1.7.d

Answer:	d. Per Tech Spec 3.5.3.b, 3 SRMs are required to be operable for core alterations. One in the quadrant where fuel is being moved and one in an adjacent quadrant. Operable SRM is required in the quadrant where the core alteration occurs and in an adjacent quadrant. The fuel assembly at core locations 28-19 is in the SE quadrant that contains operable SRM 14. It will be moved to core location 24-39, which is in the NW quadrant that contains operable SRM 12. However, an SRM in an adjacent quadrant must also be operable (either SRM 11 or SRM13)
Distractor:	a. The fuel assembly will be moved to core location 24-39, which contains operable SRM 12.
Distractor:	b. The fuel assembly at core location 28-19, contains operable SRM 14.
Distractor:	c. Per Tech Spec 3.5.3.b: SRMs are required to be operable in the quadrant where fuel is being moved and in an adjacent quadrant. All four SRMs are not required to be operable. If full core offload or reload then 4 SRMs are required.

References Provided: ~~N1-FHP-27C, and~~ core map from fuel handling lesson plan figures (01-OPS-001-101-1-02 Figure 2).

Question 15 Cross References (table item links)

10CFR55

- 43(b)(2)
- 43(b)(6)
- 43(b)(7)

Cognitive Level

- 3

Difficulty Level

- Level 4: Highest order knowledge item requiring use of problem solving skills, judgement, and maximum task complexity as well as applying procedures to determine correct answer.

NUREG 1123 KA Catalog Rev. 2

- 215004 A2.02 3.4/3.7 SRM inop condition

Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations: SRM inop condition.

Question Source

- New

PROC

- N1-FHP-27C Rev. NA

TECHSPEC

- 3.5.3.b. Rev. NA

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

16 **SYSID: 21091** **Points: 1.00**

The plant is at 100% power with N1-ST-Q5, Primary Containment Isolation Valves Operability Test, Section 8.10, Emergency Condenser Vent to Torus BV Test. Valves 05-05, EC VENT TO TORUS BV 11, and 05-07, EC VENT TO TORUS BV 12, are stroke time tested as follows:

05-05, EC VENT TO TORUS BV 11
Opened 05-05 with stroke time within IST range
Closed 05-05 with stroke time within IST range

05-07, EC VENT TO TORUS BV 12
Opened 05-07 with stroke time above IST range
Closed 05-07 with stroke time above IST range

At the completion of this test (signed off by SRO), which *one of the following* states the Tech Spec(s) entered and active in the ESL log if any?

A. TS 3.1.3. Emergency Condensers, and TS 3.3.4, Primary Containment Isolation Valves, LCO statements are met.

B. TS 3.1.3, Emergency Condensers, and TS 3.3.4, Primary Containment Isolation Valves, have active actions entered and logged.

C. TS 3.1.3, Emergency Condensers, active action(s) entered and logged. TS 3.3.4, Primary Containment Isolation Valves, LCO statement met.

D. TS 3.3.4, Primary Containment Isolation Valves, *active action*(s) entered and logged. TS 3.1.3, Emergency Condensers, LCO statement met.

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 16 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 16
System ID:	21091
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-207-1-01, EO-1.11

N1-ST-Q5, Section 10.0 Acceptance Criteria

Answer: b. If any stroke time exceeds the IST stroke time/acceptance limit or if any IST acceptance criteria is exceeded, the valve shall be declared immediately inoperable. Contact the IST Department of the inoperable valve. TS action 3.3.4.b is entered prior to performing the test and continues to apply (not exited) because of the inoperable containment isolation valve. TS 3.1.3.d applies for the inoperable EC because both the open and closed function of one valve are inoperable.

Distractor: a. Based on how far outside the IST limit, engineering could perform an operability determination and provide information to return the valves to operable status, however, valves are declared immediately inoperable when outside the IST limit until this evaluation determines they continue to be inoperable or can be declared operable.

Distractor: c. d.TS action 3.3.4.b is entered prior to performing the test and continues to apply (not exited) because of the inoperable containment isolation valve. TS 3.1.3.d applies for the inoperable EC because both the open and closed function of one valve are inoperable

References Provided: NONE

Question 16 Cross References (table item links)

10CFR55

- 43(b)(2)
- 43(b)(6)
- 43(b)(7)

Cognitive Level

- 2

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- G2.1.23 3.9/4 Ability to perform specific system and integrated plant procedures during different modes of plant operation
- 223002 PCIS/Nuclear Steam Supply Shutoff

Question Source

- New

PROC

- N1-ST-Q5 Rev. NA

TECHSPEC

- 3.1.3.D Rev. NA
- 3.3.4.b, Rev. NA

Question Setting

- C1 (License class closed reference)

17 **SYSID: 21092** **Points: 1.00**

The crew is responding to a loss of coolant accident. The RO reports TIP 4 failed to retract from the reactor core when the containment isolated on Lo-Lo RPV water level.

Per N1-OP-39, H.1.0, Securing TIP on Isolation Signal, which one of the following personnel MUST the RO consult with before placing TIP 4 SHEAR VLV switch to FIRE?

A. Reactor Engineer.

B. Chief Shift Operator.

C. Operations Manager.

D. Shift Manager.

Answer: D

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 17 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 17
System ID:	21092
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-215-1-01, EO-1.7.d

N1-OP-39, H.1.0 & 1.33

Answer:	d. Required to consult with SSS to fire the shear valve.
Distractor:	a. The Reactor Engineer would be involved with the TIP activities but is not authorized to direct activities. Can only make recommendations.
Distractor:	b. Chief Shift Operator coordinates control room actions with the other RO but cannot authorize firing the shear valve
Distractor:	c. Operations Manager would be kept informed of the containment isolation failure but the SM would make the determination to fire the shear valve and direct it.

References Provided: NONE

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 17 Cross References (table item links)

10CFR55

- 43(b)(5) X 41(b)(10)

Cognitive Level

- 1

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- 215001 A2.01 2.7/2.9 Low reactor water level: Mark-I&II(Not-BWR1)
A2. Ability to (a) predict the impacts of the following on the TRAVERSING IN-CORE PROBE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low reactor water level: Mark-1&11

Question Source

- New

PROC

- N1-OP-39 Rev. NA

Question Setting

- C1 (License class closed reference)

18 **SYSID: 21093** **Points: 1.00**

During plant operations at 100% power high temperatures have occurred in the Reactor Building. A temporary cooler has been brought into the Reactor Building Track Bay and the Inner Track Bay Door opened

When N1-ST-C5, SECONDARY CONTAINMENT AND REACTOR BUILDING EMERGENCY VENTILATION SYSTEM OPERABILITY TEST is performed to check the Reactor Building to Yard Outside Airlock Swing Away Door (Peele Door) the following data is recorded:

• Wind Speed at 30 feet	10 mph
• Rx Bldg to Atmos dP	-0.27 inches H_2O
• RBEV flow as indicated on 202-49B	1590 cfm

Regarding this specific data and Technical Specification 3.4.1 which one of the following actions is required?

A. Continue plant operations, but the test must be run again when wind speed is below 5 mph.

B. Within 4 hours close the inner door and verify reactor building leakage is restored to within Tech Spec limits.

C. Operate the temporary cooler for 3 hours to lower building temperatures then run the surveillance again.

D. Notify plant management of the test failure and initiate actions to perform a normal plant shutdown within 4 hours.

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 18 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 18
System ID:	21093
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-223-1-03, EO-1.11

Answer:	b. This test is designed to measure reactor building in-leakage and verify its <1600 cfm. The test was not successful and reactor building leakage must be established within limits within 4 hours, the inner door must be closed.
Distractor:	a. The surveillance was NOT successful, action must be taken within 4 hours to restore secondary containment.
Distractor:	c. Running the coolers may lower temperatures in the reactor building but they won't help a leaking door. Of the choices provided only b. is required.
Distractor:	d. The conditions required by T.S. 4.4.1 were NOT met but can be restored by closing the inner door, a shutdown should NOT be performed.

References Provided: Technical Specifications 3.4.1 including Figure 3.4.1

Question 18 Cross References (table item links)

10CFR55
- 43(b)(2)

Cognitive Level
- 3

DER
- NM-2004-2971, Rev. NA

Difficulty Level
- Level 4: Highest order knowledge item requiring use of problem solving skills, judgement, and maximum task complexity as well as applying procedures to determine correct answer.

NUREG 1123 KA Catalog Rev. 2
- G2.1.33 3.4/4 Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications
- 290001 Secondary CTMT

Question Source
- Bank

PROC
- N1-ST-C5 Rev. NA

TECHSPEC
- 3.4.1 Rev. NA

Question Setting
- C1 (License class closed reference)

19 **SYSID: 21094** **Points: 1.00**

With the plant at 100% power, you discover the last performance of N1-ST-M1A, Liquid Poison Pump 11 Operability Test, was completed thirty-five (35) days ago.

Which one of the following is the implication of this discovery?

A. The surveillance interval of TS SR 4.0.2 is NOT exceeded and the LCO is still met. TS SR 4.0.3 does not apply.

B. The surveillance interval of TS SR 4.0.2 is exceeded and the applicable tech spec actions must be entered. TS SR 4.0.3 does not apply.

C. TS SR 4.0.3 applies for a missed ST. The LCO must be declared not met and the applicable tech spec actions entered.

D. TS SR 4.0.3 applies for a missed ST. Declaring the LCO statement not met can be delayed up to 24 hours to perform the test.

Answer: A

Associated objective(s):

Development Area (FIO)

Question 19 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 19
System ID:	21094
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-008-362-1-03, EO-1.1

GAP-PSH-02, 4.4

Answer:	a. Although the "monthly" interval of 31days is exceeded, the surveillance interval of TS SR 4.0.2 (extension not to exceed 25%) is not exceeded. This equates to 38.75 days. The LCO is still met and adequate time exists to complete the surveillance.
Distractor:	b. The surveillance interval of 38.75 days is not exceeded. If not aware of a 25% extension, then would determine surveillance interval is exceeded and may declare the LCO not met rather than declare the surveillance missed.
Distractor:	c. TS SR 4.0.3 applies if a surveillance is missed which is not the case. The surveillance remains within the specified interval. If TS SR 4.0.3 did apply, then declaring the LCO not met can be delayed from the time of discovery up to 24 hours to perform the ST.
Distractor:	d. TS SR 4.0.3 applies if a surveillance is missed which is not the case. The surveillance remains within the specified interval.

References Provided: NONE

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 19 Cross References (table item links)

10CFR55

- 43(b)(1)

Cognitive Level

- 2

Difficulty Level

- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- G2.1.10 2.7/3.9 Knowledge of conditions and limitations in the facility license

Question Source

- New

PROC

- GAP-PSH-02 Rev. NA

TECHSPEC

- 4.0.2, Rev. NA

Question Setting

- C1 (License class closed reference)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

20 **SYSID: 21095** **Points: 1.00**

As a result of a fire in the plant which was extinguished in two (2) minutes:

- Many control room indications are lost
- The process computer has been inoperable for five (5) hours
- Per N1-OP-42, SPDS/Process Computer, appropriate requirements have been taken and/or established

Per N1-OP-42, which one of the following ADDITIONAL ACTIONS is required if eight (8) hours EXPIRES before returning the process computer to service?

A. Increase shift manning.

B. Activate the emergency plan.

C. Send an ENS report to the NRC.

D. Establish SPDES backup monitoring.

Answer: A

Associated objective(s):

Development Area (FIO)

Question 20 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 20
System ID:	21095
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-283-1-01, EO-1.8

N1-OP-42, D.1.0, D.2.0, H.2.2, H.2.4

Answer:	a. Per N1-OP-42, D.2.0 If Process Computer becomes inoperable for **greater than 8 hours**, then increase shift manning in accordance with Tech Spec 6.2.2.a.
Distractor:	b. There is insufficient criteria to determine an emergency classification is required. The fire must burn for at least 15 minutes before entering the emergency plan. If the fire was reportable, then the report would be made to the state and local agencies and the NRC no later than 62 minutes of the start of the event.
Distractor:	c. Already done – not an additional action. Per N1-OP-42, D1.0 and H.2.2: The NRC is notified in accordance with 10CFR50.72 if the Process Computer becomes inoperable for greater than four hours, coincident with a loss of a significant portion of Control Room indications.
Distractor:	d. This is a requirement if the process computer is lost, not after 8 hours. Per N1-OP-42, H.2.4: Notify Environmental Department that backup monitoring for SPDES permit requirements is necessary.

References Provided: NONE

Question 20 Cross References (table item links)

10CFR55
- 43(b)(5) ✓ also 43(b)(2)

Cognitive Level
- 1

Difficulty Level
- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- ✓ G2.1.19 3/3 Ability to use plant computer to obtain and evaluate parametric information on system or component status

Question Source
- New

PROC
- N1-OP-42 Rev. NA

Question Setting
- C1 (License class closed reference)

21 **SYSID: 21096** **Points: 1.00**

A Type 1 Procedure Change Evaluation (PCE) was generated for a Unit One (1) Operating Procedure. The Technical Verifier, QTR Review, and RPO Review are complete.

Per NIP-PRO-04, Procedure Change Evaluations and Future Procedure Enhancements, which one of the following "approvals" satisfies ALL approval requirements to consider this PCE complete?

A. SM <u>or</u> CRS approves the PCE before implementing it. No other approval is required to consider this PCE complete.

B. Manager Operations <u>or</u> GSO approves the PCE within 14 days of its performance. No other approval is required to consider this PCE complete.

C. CRS approves the PCE before implementing it <u>AND</u> GSO approves the PCE within 14 days of the CRS approval. No other approval is required to consider this PCE complete.

D. WEC SRO approves the PCE before performing it <u>AND</u> CRS approves the PCE within 14 days of the WEC SRO approval. No other approval is required to consider this PCE complete.

Answer: C

Associated objective(s):

Development Area (FIO)

Question 21 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 21
System ID:	21096
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-008-362-1-03, EO-1.9

NIP-PRO-04, 3.4.1, Item 2F, NIP-PRO-04, 3.4.4

Answer: c. Per NIP-PRO-04, 3.4.1, Item 2F: SRO review is required before implementing each Type 1 PCE. After SRO approval the PCE may be implemented.
Per NIP-PRO-04, 3.4.4: Manager, Director, or General Supervisor shall review each type 1 PCE within 14 days of SRO approval.

Distractor: a. SRO review is required before implementing each Type 1 PCE. Both the SS and CRS may perform this approval. However, Manager, Director, or General Supervisor shall review each type 1 PCE within 14 days of SRO approval. This would also be correct for a procedure change that "changes the intent" of the procedure.

Distractor: b. Manager Operations <u>or</u> GSO are both authorized to approve a Type 1 PCE within 14 days of the SRO approval, however, there is no SRO approval prior to implementing the Type 1 PCE as required.

Distractor: d. SRO review is required before implementing each Type 1 PCE and the WEC SRO is authorized to make this approval, however, the CRS is not authorized to perform the 14 day approval. Manager, Director, or General Supervisor shall review each type 1 PCE within 14 days of SRO approval

References Provided: NONE

Question 21 Cross References (table item links)

10CFR55

- 43(b)(3) ✓

Cognitive Level

- 1

DER

- NM-2004-2533, Rev. NA

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- G2.2.6 2.3/3.3 Knowledge of the process for making changes in procedures as described in the safety analysis report

Question Source

- New

PROC

- NIP-PRO-04 Rev. NA

Question Setting

- C1 (License class closed reference)

22 **SYSID: 21097** **Points: 1.00**

The phase 1 core shuffle is in progress. Which one of the following conditions meets the Fuel Handling Procedure (FHP) criteria for stopping fuel movement?

A. Last performance of the Refueling Platform Interlocks Test was completed twenty-four (24) hours ago.

B. Air bubbles are observed from the vicinity of the grapple head when a fuel assembly is latched or disengaged.

C. The fuel assembly nose piece is lowered to two (2) feet above the core top guide before establishing the correct orientation.

D. A fuel assembly is moved to the spent fuel pool and the rod block interlock light clears when the bridge is clear of the reactor core.

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 22 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 22
System ID:	21097
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-001-234-1-02, EO-1.8.b

N1-FHP-25, Attachment 4

Answer:	b. Air leakage on the fuel grapple requires stopping fuel movement. Evidence of air bubbles indicates air leakage.
Distractor:	a. This interval for this surveillance is 7 days. The surveillance is current. A similar surveillance N1-PM-SO, Refuel Platform and Grapple Inspection, is required to be completed every 12 hours.
Distractor:	c. It is acceptable to lower the fuel assembly before establishing the correct orientation, however, the correct orientation is to be established before lowering the fuel assembly into the assigned core location.
Distractor:	d. This is correct operation of the refueling interlocks. The rod block interlock light is lit when the refueling bridge is over the reactor core and the main hoist is loaded. When clear of the reactor core (proximity switch) the rod block clears.

References Provided: NONE

Question 22 Cross References (table item links)

10CFR55

- 43(b)(7)

Cognitive Level

- 2

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- G2.2.29 1.6/3.8 Knowledge of SRO fuel handling responsibilities

Question Source

- New

PROC

- N1-FHP-25 Rev. NA

Question Setting

- C1 (License class closed reference)

23 **SYSID: 21098** **Points: 1.00**

A worker has volunteered to receive an emergency exposure to SAVE A LIFE.

Per EPIP-EPP-15, Emergency Health Physics Procedure, which one of the following is the emergency dose that you can authorize as the SSS/ED without exceeding the federal limit?

A. 10 rem MINUS the workers current occupational exposure.

B. 25 rem MINUS the workers current occupational exposure.

C. 10 rem EXCLUSIVE of the workers current occupational exposure.

D. 25 rem EXCLUSIVE of the workers current occupational exposure.

Answer: D

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Question 23 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 23
System ID:	21098
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Objective: O1-OPS-004-352-1-04, EO-1.3

EPIP-EPP-15, Attachment 1

Answer:	d. Emergency exposure limits are EXCLUSIVE of the workers current occupational exposure. The limit for protecting valuable property is 10 rem and for lifesaving is 25 rem.
Distractor:	a. Emergency exposure limits are EXCLUSIVE of the workers current occupational exposure. The limit for lifesaving is 25 rem.
Distractor:	b. Emergency exposure limits are EXCLUSIVE of the workers current occupational exposure.
Distractor:	c. The limit for lifesaving is 25 rem.

References Provided: NONE

Question 23 Cross References (table item links)

10CFR55

- 43(b)(4)

Cognitive Level

- 1

Difficulty Level

- Level 2: System operation and response; requiring system/plant interrelationship knowledge to assess the situation and determine the correct answer.

NUREG 1123 KA Catalog Rev. 2

- G2.3.4 2.5/3.1 Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized

Question Source

- New

PROC

- EPIP-EPP-15 Rev. NA

Question Setting

- C1 (License class closed reference)

24 **SYSID: 21099** **Points: 1.00**

Per N1-ODP-PRO-0305, EOP/SAP Technical Bases, which one of the following describes when and why EOP-1, Attachment 2, MSIV LO/LO ISOL. BYPASS is performed with relation to performance of the step to "terminate and prevent all RPV injection except boron and CRD?"

A. Must be completed PRIOR TO this action to prevent a loss of primary containment.

B. Can be PERFORMED CONCURRENT WITH this action to prevent core wide power oscillations.

C. Must be completed PRIOR TO this action to prevent an unintended addition of heat to the torus.

D. Can be PERFORMED CONCURRENT WITH this action to prevent an unintended loss of the main condenser.

Answer: D

Associated objective(s):

Development Area (FIO)

Question 24 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 24
System ID:	21099
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00

Comment:

Objective: O1-OPS-006-344-1-03, EO-1.3

Answer: d. The MSIV low-low RPV water level isolation is bypassed since later steps in the Level branch may lower RPV water level to below the isolation setpoint. Defeating the low level isolation anticipates the possible level reduction and prevents unintended loss of the main condenser. Other MSIV isolations, such as high main steam line flow, are not defeated since they provide automatic protection for conditions requiring main steam line isolation.
The direction to bypass the isolation is prescribed as a concurrent step. The isolation may be bypassed immediately following entry of EOP-3, but lowering RPV water level to reduce reactor power or prevent power oscillations is of greater importance then maintaining the main condenser as a heat sink. Core instabilities may occur in a BWR when the reactor is operated at ~~a~~ relatively high power-to-flow ratios and recirculation flow is reduced. The potential for instabilities is largely dependent upon core inlet subcooling. The greater the subcooling, the more likely that power oscillations will occur and increase in magnitude.

Distractor: a. The direction to bypass the isolation is prescribed as a concurrent step because lowering RPV water level to reduce reactor power or prevent power oscillations is of greater importance than maintaining the main condenser as a heat sink. MSIV closure is a PCIS isolation but is not the concern. From a primary containment viewpoint the concern in this situation is the heat addition to the torus and possible HCTL challenge.

Distractor: b. The direction to bypass the isolation is prescribed as a concurrent step because lowering RPV water level to reduce reactor power or prevent power oscillations is of greater importance than maintaining the main condenser as a heat sink. The MSIV bypass is not performed to avoid power oscillations, this is avoided or minimized by lowering RPV water level to suppress reactor power.

Distractor: c. The direction to bypass the isolation is prescribed as a concurrent step because lowering RPV water level to reduce reactor power or prevent power oscillations is of greater importance than maintaining the main condenser as a heat sink. If the main condenser is available as a heat sink and the isolation is not bypassed, then heat energy would be added to the torus when RPV water level is deliberately lowered.

References Provided: NONE

Question 24 Cross References (table item links)

10CFR55
- 43(b)(5)

Cognitive Level
- 1

Difficulty Level
- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- G2.4.18 2.7/3.6 Knowledge of the specific bases for EOPs

Question Source
- New

PROC
- N1-ODP-PRO-0305 Rev. NA

Question Setting
- C1 (License class closed reference)

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

25 **SYSID: 21100** **Points: 1.00**

Given the following conditions on Unit 1:

- Emergency events result in declaration of a Site Area Emergency
- Unmonitored atmospheric release is in progress
- Wind direction from the north (0°)

Which one of the following describes your responsibility regarding Protective Action Recommendations (PARs) based on the above conditions?

A. Notify NY state to evacuate 2 miles around and 5 miles downwind of the station until the magnitude of the release is known then reassess.

B. Notify NY state to evacuate 2 miles around the station and after the magnitude of the release is known determine if evacuation 5 miles downwind is necessary.

C. Dispatch an RP Technician to perform field surveys at the site boundary near Miner road and if above general emergency levels then make protective actions recommendations.

D. Direct a Chemistry Technician to run the EDAMS computer program to determine the magnitude of the unmonitored release then determine if protective action recommendations are needed.

Answer: C

Associated objective(s):

Development Area (FIO)

Question 25 Details

Question Type:	Multiple Choice
Topic:	NRC SRO 25
System ID:	21100
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	LC1 03-01
User Text:	
User Number 1:	0.00
User Number 2:	0.00

EXAMINATION ANSWER KEY

NRC 2004 UNIT 1 SRO WRITTEN EXAMINATION

Comment:

Objective: O1-OPS-004-352-1-04, EO-1.4

EPIP-EPP-08, 3.1.1j.

Answer: c. Per EPIP-EPP-08; 3.1.1.j: IF an unmonitored atmospheric release is suspected or known to be in progress, then: Expedite the dispatch of Radiation Protection (RP) Technician. The RP Technician should be dispatched to potential plume centerline {wind direction (degrees) 180 = plume centerline}, as close to the site boundary as practical. MINER ROAD is south of the plant adjacent to the site boundary. See Attachment 1, Figure 1.4 for Site boundary location. IF readings indicate > 1 Rem/hr based on field survey perform the actions indicated in Attachment 1.

Distractor: a.b. Per EPIP-EPP-08; 3.1.1.b; PARs are not made until a General Emergency is declared regardless of the conditions present. It is not appropriate to recommend evacuations at the Site Area Emergency level even with the unmonitored release in progress. The SSS may direct a protected area evacuation or a site evacuation based on the conditions but this is not accomplished through a notification to NY state.

Distractor: d. The EDAMs computer release rate determination cannot be used to determine the magnitude of an unmonitored release. This program uses values for monitored points. Directing Chemistry to perform release rate determinations is required per the emergency plan.

References Provided: NONE

Question 25 Cross References (table item links)

10CFR55
- 43(b)(5)

Cognitive Level
- 2

Difficulty Level
- Level 3: Higher order knowledge item requiring anticipation, prediction, comprehension, and application of prior knowledge to determine the correct answer.

NUREG 1123 KA Catalog Rev. 2
- G2.4.44 2.1/4 Knowledge of emergency plan protective action recommendations

Question Source
- New

PROC
- EPIP-EPP-08 Rev. NA

Question Setting
- C1 (License class closed reference)