Name: _____

S.S. #: _____

Course: Unit 1 Reactor Operator 2005 Exam

Date: _____

Grade: _____ Graded by: _____

Training ID: _____

Trainees will be judged to have willfully violated the integrity of an examination if they are found to have:

a. Utilized unauthorized documents during the examination.

b. Secured unauthorized documents for the purpose of accessibility during an examination.

c. Solicited examination information from other trainees or any other individuals.

d. Provided examination information to other trainees during an examination.

e. Reviewed or attempted to review materials that are unauthorized, including the examination prior to implementation, the examination answer key, or the answers developed by any other trainee during the examination.

I have read and understand the above.

Signature: _____

Date: _____

This exam has been reviewed for the following:

? Validity and Inter-Rater Reliability

Exam Reviewed and Approved By: _____ _____
Instructional Technologist Date

This exam has been reviewed for the following:

- Technical consistency with the course learning objectives
- Clearly written test items
- A random sampling of learning objectives being tested
- Appropriate point values assigned each test item
- Test items written at th appropriate cognitive level for the objectives

Exam Reviewed and Approved By: _____ _____
Training Supervision Date

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

1 **SYSID: 21325** **Points: 1.00**

A power ascension following a reactor startup is in progress. Conditions are:

- 55% power on the 80 percent rod line.
- RR Pump 12 is shut down and ready for restart.

Subsequently, Reactor Recirc (RR) Pump 15 speed lowers because of a blind controller failure. Conditions are:

- Restricted Zone is entered.
- APRM indications are cycling between 38-42% on all APRMs.
- NO LPRM upscale or downscale alarms have been received.

Per N1-SOP-1.5, Unplanned Reactor Power Change, which one of the following is the correct action in response to the above conditions?

A. Insert cram rods as needed to lower reactor power

B. Withdraw control rods in sequence to raise reactor power

C. Start RR Pump 12 and raise recirc flow to raise reactor power

D. Place the Reactor Mode Switch to shutdown to scram the reactor

Answer: D

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

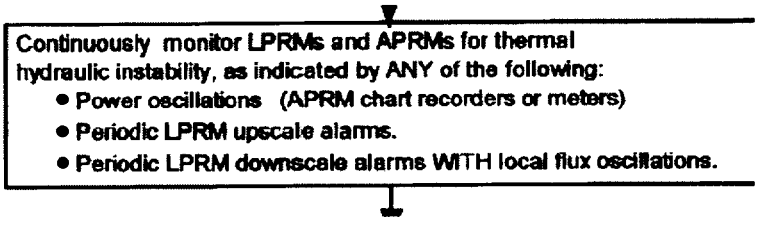
UNIT 1 REACTOR OPERATOR 2005 EXAM

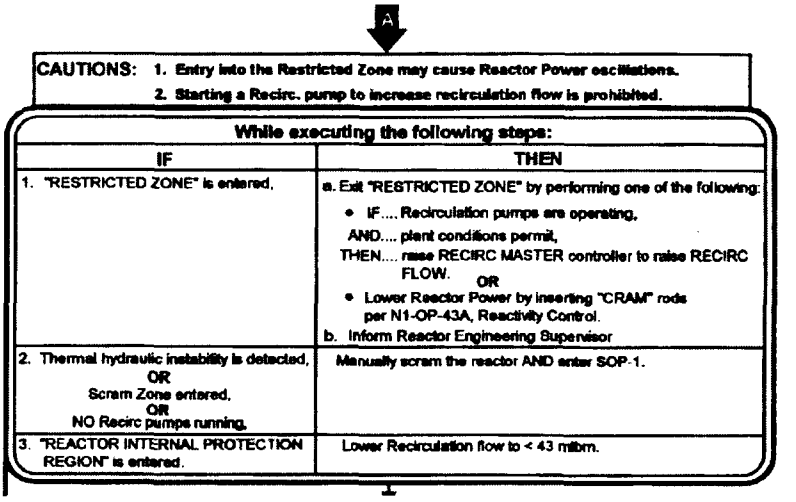
Question 1 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 1
System ID:	21325
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-202-1-01, EO-1.7g

REFERENCE:
N1-SOP-1.5
F-45683-C Sheet 1

Answer: D. is correct. The APRM indications represent THI and a manual reactor scram is required because the plant is operating in the restricted zone with THI. THI indications based on LPRM upscale and downscale alarms are not present but can be interpreted as a requirement to validate the presence of THI and lead the candidate to believe that a scram is not required at this time.





CAUTIONS: 1. Entry into the Restricted Zone may cause Reactor Power oscillations.
2. Starting a Recirc. pump to increase recirculation flow is prohibited.

While executing the following steps:

IF	THEN
1. "RESTRICTED ZONE" is entered.	a. Exit "RESTRICTED ZONE" by performing one of the following: • IF.... Recirculation pumps are operating, AND.... plant conditions permit, THEN.... raise RECIRC MASTER controller to raise RECIRC FLOW. OR • Lower Reactor Power by inserting "CRAM" rods per N1-OP-43A, Reactivity Control. b. Inform Reactor Engineering Supervisor
2. Thermal hydraulic instability is detected, OR Scram Zone entered, OR NO Recirc pumps running.	Manually scram the reactor AND enter SOP-1.
3. "REACTOR INTERNAL PROTECTION REGION" is entered.	Lower Recirculation flow to < 43 mlbm.

Distractor: **a:** This is the correct answer with no THI present. Can raise reactor power using core flow with operating RR Pumps or lower reactor power using cram rods. But evidence of THI is present requiring a reactor scram.

Distractor: **b:** With no THI present then action would be taken to exit the restricted zone. Can raise reactor power using core flow with operating RR Pumps (not by control rod withdrawal). But evidence of THI is present requiring a reactor scram.

Distractor: **c:** With no THI present then action would be taken to exit the restricted zone. Can raise reactor power using core flow with operating RR Pumps (however starting a RR pump to raise recirc flow is prohibited). But evidence of THI is present requiring a reactor scram.

Matches the K/A because the candidate must monitor nuclear instrumentation (both APRMs and LPRMs) and make determinations regarding plant control based upon this monitoring during plant conditions with a partial loss of forced core flow circulation.

REFERENCES PROVIDED TO THE CANDIDATE: NONE

(however starting a RR pump to raise recirc flow is prohibited). But evidence of THI is present requiring a reactor scram.

Matches the K/A because the candidate must monitor nuclear instrumentation (both APRMs and LPRMs) and make determinations regarding plant control based upon this monitoring during plant conditions with a partial loss of forced core flow circulation.

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 1 Cross References (table item links)

10CFR55
- 41(b)(10)
- 41(b)(7)

Cognitive Level
- 1

NUREG 1123 KA Catalog Rev. 2
- 295001 AA1.06 3.3/3.4 Neutron monitoring system

Level of Difficulty
- Level 2: System operation and response

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

Question Source
- New

Question Setting
- C1 (License class closed reference)

2 **SYSID: 21326** **Points: 1.00**

The plant is at 100% power when the following occur:

- A4-1-6, POWER BD 102 BUS VOLTAGE LOW, alarms. Nine (9) seconds later, the offsite power source available to PB 102 returned to 4160 VAC

Which one of the following describes the DESIGN response of EDG102?

A. EDG 102 is shut down and PB 102 is supplied by the offsite power source

B. EDG 102 ties to PB 102 and operates in parallel with the offsite power source

C. EDG 102 picks up PB 102 loads and the offsite power source is disconnected from PB 102

D. EDG 102 is running with its output breaker open and PB 102 is supplied by the offsite power source

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 2 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 2
System ID:	21326
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-264-1-01, EO-1.5b

Answer: C. is correct - Loss of voltage or degraded voltage on PB 102 causes an automatic start of EDG 102 and it will power PB 102. The normal AC source is automatically disconnected even if the EDG voltage and frequency have not reached rated (which takes approximately 10 seconds).

Distractor **a:** EDG 102 starts and supplies PB 102.

Distractor **b:** The normal AC source is automatically disconnected.

Distractor **d:** EDG 102 starts and supplies PB 102.

REFERENCE: N1-ARP-A4; A4-1-6, 2.a, 2.b. 2.c

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 2 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 1

NUREG 1123 KA Catalog Rev. 2

- 295003 AK2.02 4.1*/4.2* Emergency generators

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-264-1-01 Rev. NA

Question Source

- New

PROC

- N1-ARP-A4, Rev. NA

Question Setting

- C1 (License class closed reference)

3 **SYSID: 21327** **Points: 1.00**

The plant is at 100% power, in a normal alignment per the operating procedures, when the following alarm is received:

- A3 4-1 BAT. BD. 11 BATTERY BREAKER TRIP

Considering the design response before operator action is taken, which one of the following is the extent of the DC power loss, if any?

A. A partial loss of DC power to Battery Board 11 occurs; not a complete loss

B. A complete loss of DC power to Battery Board 11 occurs; not just a partial loss

C. MG 167 maintains DC power to Battery Board 11 without a loss of DC power

D. Charger 161A maintains DC power to Battery Board 11 without a loss of DC power

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 3 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 3
System ID:	21327
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-263-1-01, EO-1.8b

Answer: B. is correct - Battery Board 11 DC power is lost because the battery is unavailable and the SBC (Static Battery Charger) DC output breaker trips open.

Per N1-OP-47A, D.11.0: The SBC DC breaker is interlocked to trip open whenever the main battery breaker is open.

Distractor: A: is incorrect - Complete loss of power to BB11 occurs.

Distracter: C: is incorrect - SBC DC output breaker trips open. MG 167 is non safety-related but can be used as a battery charger.

Distracter: D: is incorrect - SBC DC output breaker trips open.

REFERENCE: ARP A3 4-1, N1-OP-47A B.4.0, D.11.0

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 3 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 1

NUREG 1123 KA Catalog Rev. 2

- 295004 AA2.02 3.5/3.9 Extent of partial or complete loss of D.C. power

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-263-1-01 Rev. NA

Question Source

- New

PROC

- N1-ARP-A3 Rev. NA
- N1-OP-47A Rev. NA

Question Setting

- C1 (License class closed reference)

4 **SYSID: 21328** **Points: 1.00**

The Unit is operating at 100% when the Main Turbine trips without scramming the reactor. The Main Turbine Bypass Valves fail to open.

Which one of the following is the correct RPS ATWS–RPT and ATWS-ARI response to the above conditions?

A. RRP drive motor breakers and M-G Set field breakers open, ARI valves open

B. RRP drive motor breakers and M-G Set field breakers open, ARI valves remain closed

C. RRP drive motor breakers and M-G Set field breakers remain closed, ARI valves remain closed

D. RRP drive motor breakers open but the M-G Set field breakers remain closed, ARI valves open

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 4 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 4
System ID:	21328
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-212-1-01, EO-1.5b

Answer: A. is correct - A two-out-of-two-taken-once logic is used, such that a high reactor pressure (1135 psig instantaneous – no time delay) or a Lo-Lo reactor level (+5 inches with a 9 ± 1 second TD) in both sensors in the sample RPS channel will cause ATWS-RPT to initiate.
The ATWS-RPT system utilized an energize-to-function 125 VDC circuit to:
1. Energize trip coils located in each of the recirculation pump M-G set field breaker cabinets, causing the field breakers to open and consequent loss of power to the pump motors, and
2. Trip the RRP drive motor breakers.
This causes the RRPs to trip with minimal coast down which reduces flow through the reactor core to the natural circulation condition. This creates increased voiding in the core, thus reducing power. The reduced flow in the core may cause reactor power oscillations.

The ATWS/RPT initiation on high reactor pressure is instantaneous. However, there is no seal-in function associated with Lo-Lo Reactor Level ATWS-RPT initiation. If the Lo-Lo Reactor Level initiation signals clear prior to the nine (9) second time delay, the recirc pumpswill not trip. Once tripped, a normal pump start sequence may be accomplished provided no initiating signals are present.

The logic to initiate ATWS-ARI come from the same sensors that initiate the ATWS-RPT. A two-out-of-two-

taken-once ATWS logic is used, such that a high reactor pressure (1135 psig) or a Lo-Lo reactor level (+5 in.) in both sensors in the same logic channel will initiate ATWS-ARI. This system utilizes energize-to-function 125 volt DC solenoid operated valves for the purpose of depressurizing the Scram Air Header. A failure of the DC power supply will not cause the system to actuate, nor allow the solenoid valves to vent. DC power can be transferred from one battery board to the other locally at ARI CABINET #3 (Rx. Bldg. 237 HCU module area). There are no time delays associated with the actuation of ATWS-ARI. When the logic is initiated, a seal-in function prevents resetting of ATWS-ARI until a 25 second timer resets, even if the initiating signals clear. This ensures that the reactor scram function is completed.

Distractor: B: is not correct - ARI valves also open; they do not remain closed.

Distractor: C: is not correct - RRP drive motor breakers and M-G Set field breakers open, they do not remain closed. ARI valves open; they do not remain closed.

Distractor: D: M-G Set field breakers open, they do not remain closed.

REFERENCE: N1-OP-40 Section B

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 4 Cross References (table item links)

10CFR55

- 41(b)(5)
- 41(b)(6)

Cognitive Level

- 2

NUREG 1123 KA Catalog Rev. 2

- 295005 AA2.04 3.7/3.8 Reactor pressure

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-212-1-01 Rev. na

Question Source

- New

PROC

- N1-OP-40 Rev. NA

Question Setting

- C1 (License class closed reference)

5 **SYSID: 21329** **Points: 1.00**

The plant is at 100% with 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
- FW LVL SP SETDN INIT light is OFF
- FW flow is 1.9×10^6 on the Motor Driven pumps

In response to the above conditions, the CSO set the FEEDWATER MASTER CONTROLLER at +55 inches. Which one of the following is the reason for this action?

A. Control **FCV #11** injection in response to the initial **increase** in core voids

B. Control **FCV #13** injection in response to the initial **increase** in core voids

C. Control **FCV #11** injection in response to the initial **decrease** in core voids

D. Control **FCV #13** injection in response to the initial **decrease** in core voids

Answer: D

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 5 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 5
System ID:	21329
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-259-1-02, EO-1.5b

Answer: D. is correct - Set point set down has failed and the CSO action per the reactor scram procedure achieves the same result as if the actual set point set down circuit had operated correctly. Set point set down, or in this case the operator action to achieve set point set down prevents FCV #13 from responding to the initial collapse (reduction in core voids) by opening and allowing more water than is required to flow into the core. This results in a more controlled increase in water level during the initial level recovery period following a scram.

Distracter: A: is not correct - FCV #13, not FCV#11 is the valve controlled in the set point set down. Additionally, for the plant conditions specified, FCV #11 is in manual and FCV#13 is in automatic and responds to signal from the master flow controller. When the master flow controller is adjusted to +55 inches, FCV#13 is being controlled not FCV#11. In response to a reactor scram, the core voids collapse which is a reduction in core voids not an increase in core voids.

Distracter: B: is not correct - In response to

a reactor scram, the core voids collapse which is a reduction in core voids not an increase in core voids.

Distracter: C: FCV #13, not FCV#11 is the valve controlled in the set point set down. Additionally, for the plant conditions specified, FCV #11 is in manual and FCV#13 is in automatic and responds to signal from the master flow controller. When the master flow controller is adjusted to +55 inches, FCV#13 is being controlled not FCV#11.

Question matches the K/A because the operator action taken is the same action accomplished automatically by the set point set down if it occurred automatically (had not failed) and the candidate must state the reason the action to reduce the master level controller set point to + 55 inches was taken.

REFERENCE: N1-OP-16, B.3.0

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 5 Cross References (table item links)

10CFR55

- 41(b)(1)
- 41(b)(10)
- 41(b)(7)

Cognitive Level

- 1

NUREG 1123 KA Catalog Rev. 2

- 295006 AK3.04 3.1/3.3 Reactor water level setpoint setdown: Plant-Specific

Level of Difficulty

- Level 2: System operation and response

LP

- O1-OPS-001-259-1-02 Rev. na

Question Source

- New

PROC

- N1-OP-16 Rev. NA

Question Setting

- C1 (License class closed reference)

6 **SYSID: 21330** **Points: 1.00**

With the plant at 100% power, a control room evacuation is directed. The immediate operator actions were not taken prior to leaving the control room.

If personnel assume their assigned roles during the control room evacuation as required per N1-SOP-21.2, Control Room Evacuation, which one of the following describes who is expected to perform the actions to manually scram the reactor from outside the control room?

A. SM

B. CSO

C. In Plant E

D. Control Room E

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 6 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 6
System ID:	21330
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-006-342-1-01, EO-1.2b

Answer: B. is correct - Per N1-SOP-21.2, Flowpath B, actions for ASSS/CSO, if the reactor was not scrammed prior to leaving the control room:
1. Proceed to RSP 12 on TB 277 AND place MG 141 switch in TRIP position.
2. Proceed to RSP 11 on TB 250 AND place MG 131 switch in TRIP position.
3. Confirm CONTROL RODS IN white light lit on RSP 11 OR 12.
4. Inform SSS Reactor has been scrammed.

Matches K/a in that it tests not only where specific actions are performed to accomplish a task outside the control room but also who is expected to perform the actions. In this case, the candidate must not only determine how to scram the reactor from outside the control room but must also demonstrate an understanding of the assigned staffing for control room evacuation and their association to the control stations where the actions are taken to perform the task specified. This also raises the cognitive level from memory to higher order.

Distracter: A: is not correct - SM. Actions are outlined in PATH A. Proceeds to the TSC to assume the role as the ED. Is informed of the manual scram by the CSO when the actions are complete.

Distracter: C: is not correct - In-Plant E. Actions are outlined in PATH D. Responsible for manual vessel isolations and manual FW FCV operation, but does not take the

actions to manually scram the reactor.

Distracter: D: is not correct - Control Room E. Actions are outlined in PATH C. Responsible for tripping the main turbine and disengaging FWP 13, but does not take the actions to manually scram the reactor.

REFERENCE:
Per N1-SOP-21.2, Flowpath B,

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 6 Cross References (table item links)

10CFR55
- 41(b)(10)
- 41(b)(6)

Cognitive Level
- 2

NUREG 1123 KA Catalog Rev. 2
- 295016 AK2.02 4.0*/4.1* Local control stations: Plant-Specific

Level of Difficulty
- Level 2: System operation and response

LP
- O1-OPS-006-342-1-01 Rev. na

Question Source
- New

PROC
- N1-SOP-21.2, Rev. NA

7 **SYSID: 21331** **Points: 1.00**

With the plant at 100% power, a control room evacuation is directed. The immediate operator actions were completed prior to leaving the control room. At Remote Shutdown Panel 11:

- EC 11 and 12 were not initiated
- CHANNEL 11 CONTROL TRANSFER keylock switch is in EMERG
- EC Steam Supply IV 39-07R and 39-09R are verified open
- EC 111/112 LEVEL CONTROL TRANSFER switch is in LOCAL

In accordance with N1-SOP-21.2, Control Room Evacuation, which one of the following describes the desired control and monitoring for the RPV cool down under the above conditions?

A. Cycle EC Steam Supply IV (39-09R) and ensure EC condenser shell level is being maintained automatically by monitoring this parameter at the RSP

B. Cycle EC Condensate Return IV (39-05) and ensure EC condenser shell level is being maintained automatically by monitoring this parameter at the RSP

C. Cycle EC Steam Supply IV (39-09R) and ensure EC condenser shell level is maintained between 6' and 7' by the NAOC providing manual makeup at EC 11

D. Cycle EC Condensate Return IV (39-05) and ensure EC condenser shell level is maintained between 6' and 7' by the NAOC providing manual makeup at EC 11

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 7 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 7
System ID:	21331
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-006-342-1-01, EO-1.2b

Answer: B. is correct - Per N1-SOP-21.2, Flowpath B:
Commence a normal RPV cooldown (<100°F/hr) with EC 11:
1. Place CHANNEL 11 CONTROL TRANSFER keylock switch in EMERG position to transfer control to RSP.
2. Verify open 39-07 AND 39-09.
3. Control RPV cooldown by cycling open and closed 39-05, EMERGENCY CONDENSER COND RTN IV 11.

Control EC CONDENSER SHELL level between 6' and 7':
1. Place EC 111/112 LEVEL CONTROL TRANSFER switch to LOCAL.
2. Verify AUTO control by observing "A" on status panel (normally selected to A).
NOTE: If instrument air was lost and 60-17 failed, the required action would be to manually throttle closed 60-12, BV for EC 11 Makeup LCV, to control level.

Per N1-SOP-21.2, 5.2:
Parameters monitored at each remote shutdown panel (RSP) include analog indications for Emergency Condenser water level.

Matches K/A in that the question requires the candidate to both (1) correctly **operate** the EC to establish cool down given the Local Transfer Switch actions and EC not having initiated, and the automatic actions having been completed prior to leaving the control room, and (2) correctly indicate how to **monitor** and maintain EC condenser shell side level when the cool down is established.

Distracter: A: is not correct - If the EC had auto initiated (>1080 psig for 12 seconds) before control was taken at RSP 11, operation of the EC Condensate Return valve (the desired means) would be unavailable and the only method to reduce cool down rate is to throttle on the steam supply. However, because the EC have not initiated, the desired means (EC Condensate Return Valve) is available.

Distracter: C: is not correct - If the EC had auto initiated (>1080 psig for 12 seconds) before control was taken at RSP 11, operation of the EC Condensate Return valve (the desired means) would be unavailable and the only method to reduce cool down rate is to throttle on the steam supply. However, because the EC have not initiated, the desired means (EC Condensate Return Valve) is available. Under these conditions EC condenser shell level control is automatic and its correct operation is determined by monitoring EC level at the RSP. If instrument air was lost and 60-17 failed, the required action would be to manually throttle closed 60-12, BV for EC 11 Makeup LCV, to control level.

Distracter: D: is not correct - Under these conditions EC condenser shell level control is automatic and its correct operation is determined by monitoring EC level at the RSP. If instrument air was lost and 60-17 failed, the required action would be to manually throttle closed 60-12, BV for EC 11 Makeup LCV, to control level.

REFERENCE:
Per N1-SOP-21.2, Flowpath B, and Section 5.2.

REFERENCES PROVIDED TO THE CANDIDATE:
NONE

Question 7 Cross References (table item links)

10CFR55

- 41(b)(10)
- 41(b)(7)

Cognitive Level

- 2

NUREG 1123 KA Catalog Rev. 2

- 295016 AA1.09 4/4 Isolation/emergency condenser(s): Plant-Specific

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-006-342-1-01 Rev. na

Question Source

- New

PROC

- N1-SOP-21.2, Rev. NA

Question Setting

- C1 (License class closed reference)

8 **SYSID: 21332** **Points: 1.00**

With the plant in a refueling outage with reactor vessel re-assembly in progress, a loss of RBCLC occurs. The CRS directs increased temperature monitoring for MAJOR system loads.

Which one of the following RBCLC loads requires increased temperature monitoring based upon the CRS direction?

A. Drywell Air Coolers

B. Control Room Ventilation

C. Fuel Pool Heat Exchangers

D. Drywell Equipment Drain Tank Coolers

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 8 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 8
System ID:	21332
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-208-1-01, EO-1.7f

Answer: C: is correct - This is a major load for the shutdown condition. The candidate must discriminate between minor and major system loads for the cold shutdown condition with refueling completed – larger heat load in the spent fuel pool.

Distracter: A: is not correct - This is a minor, not a major, cold shutdown load.

Distracter: B: is not correct - This is a minor, not a major, cold shutdown load.

Distracter: D: is not correct - This is a minor, not a major, cold shutdown load.

REFERENCE:
N1-SOP-11.1, Table 11.1

Question matches the K/A since the candidate is required to determine which RBCLC load requires increased monitoring when a complete loss of RBCLC occurs. The candidate must discriminate between minor and major system loads for the cold shutdown condition with refueling completed – larger heat load in the spent fuel pool. Under the current conditions there are no system loads to operate yet so evaluating the "monitor" rather than "operate" part of this K/A has the greater importance.

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 8 Cross References (table item links)

10CFR55

- 41(b)(4)

Cognitive Level

- 1

NUREG 1123 KA Catalog Rev. 2

- 295018 AA1.02 3.3/3.4 System loads

Level of Difficulty

- Level 2: System operation and response

LP

- O1-OPS-001-208-1-01 Rev. na

Question Source

- New

PROC

- N1-SOP-11.1, Rev. NA

Question Setting

- C1 (License class closed reference)

9 **SYSID: 21333** **Points: 1.00**

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

- L1-3-7, INST AIR SYSTEM, alarms
- A193 IA DRYERS OR FILTERS_OFFN prints out on the alarm typer
- An operator dispatched to the local panel reports INST AIR DRYERS DISCH PRESS LOW
- NO automatic actions have occurred

Which one of the following is the correct action per ARP L1-3-7?

A. Both 94-164 and 94-206 must be opened. Opening only 94-164 or 94-206 will NOT correct the condition

B. Either 94-164 or 94-206 must be opened. Opening both 94-164 and 94-206 is NOT required to correct the condition

C. Open 94-164 and if it can't be opened then open 94-38. Opening 94-206 is NOT an action that will correct the condition

D. Open 94-206 and if it can't be opened then open 94-38. Opening 94-164 is NOT an action that will correct the condition

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 9 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 9
System ID:	21333
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-278-1-01, EO-1.7e

Answer: B. is correct - If **INST AIR DRYERS HIGH DIFF PRESSS** or **INST AIR DRYERS DISCH PRESS LOW** in alarm, then perform the following:
Verify auto bypass 94-164 open AND high dp/low discharge pressure condition clear on DPIS 113-476/PS 113-474. *(corrective action)*
Verify auto bypass 94-206 open on low discharge pressure. *(corrective action)*
If 94-164 OR 94-206 cannot be opened, THEN open 94-38. *(alternate action)*

Distracter: A: is not correct - Opening either valve will correct the condition. Both valves are not required to be opened.

Distracter: C: is not correct - 94-38 is an alternate valve to open performed only if the primary actions (opening 94-164 or 94-206) cannot be completed. Opening 94-206 will correct the condition.

Distracter: D: is not correct - 94-38 is an alternate valve to open performed only if the primary actions (opening 94-164 or 94-206) cannot be completed. Opening 94-164 will correct the condition.

Question matches the K/A because the valve manipulations to be determined by the candidate are

specified in the ARP actions.

REFERENCE: N1-ARP-L1; L1-3-7; C-18011-C, Sheet 2

REFERENCES PROVIDED TO THE CANDIDATE: C-18011-C, Sheet 1 and 2

Question 9 Cross References (table item links)

10CFR55
- 41(b)(7)

Cognitive Level
- 2

DRW
- C-18011-C Rev. NA

NUREG 1123 KA Catalog Rev. 2
- G2.4.31 3.3/3.4 Knowledge of annunciators alarms and indications, and use of the responseinstructions
- 295019 Partial or Complete Loss of Instrument Air

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-001-278-1-01 Rev. na

Question Source
- New

PROC
- N1-ARP-L1 Rev. NA

Question Setting
- C1 (License class closed reference)

10 **SYSID: 21334** **Points: 1.00**

The plant is shutdown preparing for a refueling outage. Conditions are:

- RPV level at +85 inches
- SDC 11 is in service
- Reactor Recirculation Pump (RRP) 14 operating
- All other Reactor Recirculation Loops are isolated
- A seismic event occurs; SDC Pump 11 trips and RRP 14 trips

Which one of the following recovery actions is effective in avoiding thermal stratification?

A. One RRP in operation. NO SDC loop system in service. RPV water level can remain at its current level

B. One SDC loop system in operation. NO RRP in operation. RPV water level can remain at its current level

C. RPV water level is raised above the MSL nozzles. NO RRP in operation. No SDC system in operation. RRP 14 is isolated

D. RPV water level is raised to the top of the steam separator. NO RRP in operation. One SDC system in operation. RRP 14 is isolated

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 10 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 10
System ID:	21334
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-205-1-01, EO-1.6a, EO-1.6b

Answer: A. is correct - N1-OP-4, D.8.0: To prevent thermal stratification while shutdown with fuel in the Reactor Vessel, forced circulation through the Core shall be maintained with at least one Recirc Pump running or Vessel Level shall be maintained above the Main
Steam Line Nozzles with the Shutdown Cooling system in service and all Recirc Pump Suction, or Discharge and Discharge Bypass Valves closed.

Distracter: B: is not correct - With SDC in operation and NO RRP running, level must be raised to above the elevation of the MSL nozzles and RRP 14 isolated to prevent thermal stratification.

Distracter: C: is not correct - SDC must also be in operation to prevent thermal stratification.

Distracter: D: is not correct - Level must be raised above the elevation of the MSL nozzles. The top of the steam separator is at an elevation below the elevation of the MSL nozzles and is too low to prevent thermal stratification.

REFERENCE: N1-OP-4, D.8.0

REFERENCES PROVIDED TO THE CANDIDATE:

NONE

Question 10 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 1

NUREG 1123 KA Catalog Rev. 2

- 295021 AK1.02 3.3/3.4 Thermal stratification

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-205-1-01 Rev. na

Question Source

- New

PROC

- N1-OP-4 Rev. NA

Question Setting

- C1 (License class closed reference)

11 **SYSID: 21335** **Points: 1.00**

During a refueling outage, which one of the following requires notification of plant personnel to evacuate BOTH the Reactor Building EL 340' AND the drywell per N1-SOP-6.1, Loss of SFP/Rx Cavity Level/Decay Heat Removal, assuming the Emergency Plan is NOT entered?

A. The running Shutdown Cooling pumps trip and cannot be immediately restarted during core offload

B. During a fuel shuffle the Spent Fuel Pool and Reactor Cavity water level are discovered to be two inches below the normal water level

C. The Refueling Bridge high range ARM alarms as an irradiated fuel assembly that is being removed from the reactor core approaches normal-up

D. The Reactor Building East Wall Elevation 340 ARM alarms following a loss of Spent Fuel Pool Cooling

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 11 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 11
System ID:	21335
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-006-342-1-01, EO-1.1, EO-1.2b

Answer:	C: is correct - Per N1-SOP-6.1: Evacuation of personnel from both the 340' elevation and the Drywell are required if an irradiated Fuel Bundle has been uncovered or if the Refuel Bridge High Range ARM Alarm has sounded. Other conditions require evacuation of non-essential personnel from the 340' but not the drywell and these conditions are represented in the distracters.
Distracter:	A: is not correct - If RPV decay heat removal capability is lost, flow path B is taken and no evacuations are required.
Distracter:	B: is not correct - If fuel pool inventory or reactor cavity inventory is lowering, an evacuation of non-essential personnel from the 340' elevation is required. All personnel are not evacuated and drywell evacuation is not required.
Distracter:	D: is not correct - ARM 340 East Wall requires a RB 340 evacuation (ARP-H1 4-8), but not a Drywell evacuation. This ARM is also not referred to in SOP-6.1. SInce the Emergency Plan is not entered, as provided in the question, candidate cannot

argue that a complete reactor building evacuation is warranted.

Question matches the K/A because the candidate must assess system conditions during refueling operations (accident conditions) and determine at which point evacuation of all personnel is required from certain locations (evacuation is performed through notification by announcement).

REFERENCE:
N1-SOP-6.1

REFERENCES PROVIDED TO THE CANDIDATE:
NONE

Question 11 Cross References (table item links)

10CFR55
- 41(b)(12)
- 41(b)(13)

Cognitive Level
- 1

NUREG 1123 KA Catalog Rev. 2
- G2.1.14 2.5/3.3 Knowledge of system status criteria which require the notification of plant personnel
- 295023 Refueling Accidents

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-006-342-1-01 Rev. na

Question Source
- New

PROC
- N1-SOP-6.1, Rev. NA

Question Setting
- C1 (License class closed reference)

12 **SYSID: 21336** **Points: 1.00**

Following a small LOCA in the drywell the following events occur at the noted times:

- 02:18 - Venting primary containment through RBEVS initiated
- 02:47 - The reactor is manually scrammed
- 02:51 - Drywell pressure spikes to 5.2 psig and stable
- 03:03 - RPV water level is at +20 inches and stable

Which one of the following actions is required, based on the above conditions?

A. Manually secure primary containment venting

B. Restore the reactor coolant sampling capability

C. Perform rapid Cleanup start for RPV level control

D. Restore containment H2-O2 monitoring capability

Answer: D

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 12 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 12
System ID:	21336
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-223-1-01, EO-1.4c

Answer: D: is correct - Containment isolation has occurred. Vessel isolation has not occurred.

Distractor: A: is not correct - Automatically isolated on containment isolation at 3.5 psig in the drywell. It does not need to be manually secured.

Distractor: B: Sample valves are still open. They isolate on a vessel isolation which occurs at +5 inches, and do not isolate on containment isolation.

Distractor: C: Cleanup isolates on vessel isolation and not containment isolation. This action is taken following cleanup isolation to restore cleanup to lower reactor water level, not to raise it.

Question matches the K/A because the candidate must associate the conditions to a containment isolation present on high drywell pressure, and a vessel isolation not present, and use this information to determine the PCIS/NSSSS valves that are affected. As a result of this determination, the candidate must determine which of the procedure actions are appropriate and required based on the conditions presented.

REFERENCE: N1-SOP-40.2, N1-OP-9 H.3.0

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 12 Cross References (table item links)

10CFR55

- 41(b)(9)
- 41(b)(79)

Cognitive Level

- 2

NUREG 1123 KA Catalog Rev. 2

- 295024 EK2.07 3.9/3.9 PCIS/NSSSS

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-223-1-01 Rev. na

Question Source

- New

PROC

- N1-OP-9 Rev. NA
- N1-SOP-40.2, Rev. NA

Question Setting

- C1 (License class closed reference)

13 **SYSID: 21337** **Points: 1.00**

Given the following conditions during an ATWS:

- Both ECs are in service, the main condenser is unavailable
- Reactor power is 20%
- Torus water level is 11.0 feet
- Torus water temperature is 130°F
- Liquid Poison has been initiated

Which one of the following is the HIGHEST reactor pressure permissible without violating the Heat Capacity Temperature Limit?

A. 960 psig

B. 860 psig

C. 760 psig

D. 660 psig

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 13 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 13
System ID:	21337
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-006-344-1-03, EO-1.2

Answer: B. is correct - From the reactor pressures presented, the highest reactor pressure that is still within the HCTL is 860 psig. Because the torus water level is within the TS limit, Curve A is used and not curve B. If Curve B is incorrectly used, then from the reactor pressures presented, the highest reactor pressure that is still within the HCTL is 660 psig.

Distractor: A: is not correct - Above the HCTL.

Distractor: C: is not correct - Below the HCTL if the Curve A is used but is not the highest reactor pressure permissible of those listed.

Distractor: D: is not correct - Below the HCTL Curve B and is the only reactor pressure below this curve. Could be chosen if the incorrect Curve B is used.

REFERENCE: EOP Figure M.

REFERENCES PROVIDED TO THE CANDIDATE: N1-EOP-3, N1-EOP-4

Question 13 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 3

NUREG 1123 KA Catalog Rev. 2

- 295025 EA2.01 4.3*/4.3. Ability to determine/interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor pressure

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-006-344-1-03, Rev. NA

Question Source

- New

PROC

- N1-EOP-3 Rev. NA

Question Setting

- C1 (License class closed reference)

14 **SYSID: 21338** **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 Water Temperatures are as follows:

- 08:00, at 82°F at the start of the test.
- 08:10, at 85°F
- 08:15, at 86°F
- 08:25, at 90°F
- 08:30, at 91°F
- 08:35, at 93°F when the test is completed

Which one of the following is the Technical Specification implication of the above conditions?

The Torus Average Water Temperature

A. REMAINED WITHIN the TS limit during the performance of the test (08:00 to 08:35).
N1-ST-M3, SUPPRESSION CHAMBER - DRYWELL RELIEF VALVES EXERCISING, is required to be performed in response to this test

B. REMAINED WITHIN the TS limit during the performance of the test (08:00 to 08:35).
N1-ST-M3, SUPPRESSION CHAMBER - DRYWELL RELIEF VALVES EXERCISING, is NOT required to be performed in response to this test

C. EXCEEDED the TS limit during actual performance of the test at 08:15.
N1-ST-M3, SUPPRESSION CHAMBER - DRYWELL RELIEF VALVES EXERCISING, is required to be performed in response to this test

D. EXCEEDED the TS limit during actual performance of during the test at 08:30.
N1-ST-M3, SUPPRESSION CHAMBER - DRYWELL RELIEF VALVES EXERCISING, is NOT required to be performed in response to this test

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 14 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 14
System ID:	21338
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-223-1-02, EO-1.11a, EO-1.11d

Answer: A: is correct - 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. Twenty-four hours are permitted to return the temperature to less than 85°F. Per TS 3.3.6/4.3.6.a: Suppression Chamber - Drywell Vacuum Relief Valves testing shall be performed monthly during all periods when this system is required, **or following the release of energy to the suppression chamber**, or following maintenance on the valves or associated equipment.

Distractor: B: is not correct - Per TS 3.3.6/4.3.6.a: Suppression Chamber - Drywell Vacuum Relief Valves testing shall be performed monthly during all periods when this system is required, **or following the release of energy to the suppression chamber**, or following maintenance on the valves or associated equipment.

Distractor: C: is not correct - 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. This limit was not exceeded between 08:00 and 08:35.

Distractor: D: is not correct - 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. This limit was not exceeded between 08:00 and 08:35. Regardless of whether the TS limit for torus water temperature is exceeded or not, per TS 3.3.6/4.3.6.a: Suppression Chamber - Drywell Vacuum Relief Valves testing shall be performed monthly during all periods when this system is required, **or following the release of energy to the suppression chamber**, or following maintenance on the valves or associated equipment.

Matches K/A because the ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications as related to high suppression pool temperature is tested by knowing that the normal limit of 85°F is adjusted by 10°F during testing that adds heat to the torus and determining that the limit was not exceeded. Additionally, Because of the heat energy to the tours, the vacuum breakers surveillance is required to be performed.

REFERENCE:
TS 3.3.2.d, TS 3.3.6/4.3.6.a

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 14 Cross References (table item links)

10CFR55

- 41(b)(10)
- 41(b)(7)

Cognitive Level

- 2

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
- 295026 Suppression Pool High Water Temperature

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-223-1-02 Rev. NA

Question Source

- New

TECHSPEC

- TS 3.3.2.d, Rev. NA
- TS 3.3.6/4.3.6a, Rev. NA

Question Setting

- C1 (License class closed reference)

15 **SYSID: 21339** **Points: 1.00**

Given the following plant conditions:

- Drywell temperature is 298°F and rising slowly
- Drywell pressure is 6 psig and rising slowly
- Torus temperature is 80°F and steady
- Torus Pressure is 5 psig and rising slowly
- RPV pressure is 875 psig and steady
- RPV level is 75 inches and steady

Which one of the following actions is required by EOPs, based on the above conditions?

A. Initiate containment sprays and if ineffective in reducing drywell temperature then depressurize the reactor

B. Immediately perform an RPV blow down because containment pressure does not permit sprays at this time

C. Immediately vent the primary containment from the drywell because containment pressure does not permit sprays at this time

D. Initiate containment sprays and if ineffective in reducing drywell temperature then vent the primary containment from the drywell

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 15 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 15
System ID:	21339
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-006-344-1-04, EO-1.2, EO-1.3

Answer: B: is correct - Drywell temperature cannot be restored and maintained below 300°F; containment sprays are not permitted because operation is within the NO SPRAY region of the containment spray initiation limit (Figure K). RPV blowdown is required at this time.

Distractor: A: is not correct - Containment sprays are not permitted because operation is within the NO SPRAY region of the containment spray initiation limit (Figure K).

Distractor: C: is not correct - RPV blowdown is required not venting. PCPL is not being challenged and therefore venting is not the correct action.

Distractor: D: is not correct - Containment sprays are not permitted because operation is within the NO SPRAY region of the containment spray initiation limit (Figure K). PCPL is not being challenged and therefore venting is not the correct action.

Question matches the K/A because the candidate must monitor the containment parameters and monitor their association to the containment spray initiation limit to determine containment sprays are not appropriate although

they are required in response to the high drywell temperature.

REFERENCE:
N1-EOP-4, Figure K, DW/T-4, DW/T-6.

REFERENCES PROVIDED TO THE CANDIDATE: N1-EOP-4

Question 15 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 3

NUREG 1123 KA Catalog Rev. 2
- 295028 EA1.01 3.8/3.9 Drywell spray: Mark-I&II

Level of Difficulty
- Level 4: Highest order knowledge

LP
- O1-OPS-006-344-1-04 Rev. na

PROC
- N1-EOP-4 Rev. NA

Question Setting
- C1 (License class closed reference)

16 **SYSID: 21400** **Points: 1.00**

Following a seismic event at 100% power, a torus leak concurrent with a LOCA occurs:

- Reactor water level is at -20 inches and lowering 10 inches/minute
- Torus water level is at 9.0 feet and lowering 3 inches/minute following initiation of Torus fill
- The CRS directs an RPV Blowdown because of the lowering torus water level

Which one of the following is the reason for the RPV Blowdown directed by the CRS?

A. Ensure the containment vent valves can be operated for containment heat rejection and venting

B. Ensure opening an ERV will not result in exceeding the capability of its tailpipe, quencher, or associated supports

C. Ensure the containment water volume will be adequate for the ECCS to flood the reactor vessel following a large break LOCA

D. Ensure continued plant operation does not result in exceeding the maximum pressure capability of the primary containment

Answer: D

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 16 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 16 REV 1
System ID:	21400
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-006-344-1-04, EO-1.3

Answer: D: is correct - Low Torus water level will result in direct pressurization of the Torus air space if the discharge holes are uncovered (at 8 feet) and result in exceeding the Torus design pressure.

Distractor: A: is incorrect - Concern for rising Torus level, not lowering.

Distractor: B: is incorrect - Concern for high Torus water level, not a low Torus water level.

Distractor: C: is incorrect - At 8 feet in the Torus, there is still sufficient volume for reflood.

Question matches the K/A because the candidate must select the correct reason for performing an RPV Blowdown (emergency depressurization) based on a low torus water level (suppression pool water level).

Question 16 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 2

NUREG 1123 KA Catalog Rev. 2

- 295030 EK3.01 3.8/4.1 Emergency depressurization

Level of Difficulty

- Level 2: System operation and response

LP

- O1-OPS-006-344-1-04 Rev. na

OTHER_REFS

- (REPORT) NER-1M-095, Rev. NA

Question Source

- New

PROC

- N1-EOP-4 Rev. NA

Question Setting

- C1 (License class closed reference)

17 **SYSID: 21341** **Points: 1.00**

Given the following conditions:

- The plant has experienced a LOCA
- A loss of ALL injection has occurred
- RPV water level is currently –115 inches (actual level) and stable

Based on the above conditions, which one of the following is the CURRENT STATE of adequate core cooling?

A. ASSURED through steam cooling

B. ASSURED through core submergence

C. LOST but is restored when level reaches -109" after RPV blowdown

D. LOST but is restored when level reaches -84" after RPV blowdown, not -109"

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 17 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 17
System ID:	21341
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-006-344-1-02, EO-1.3

Answer: A: is correct - ACC through steam cooling per N1-EOP-9 while RPV level is maintained above -121 inches. Below -121 inches ACC is lost requiring RPV blowdown and low pressure injection to restore RPV level and reestablish ACC.

Distractor: B: is not correct - ACC through submergence is employed when RPV level is maintained above TAF (-84"). The current mechanism is steam cooling.

Distractor: C: is not correct - ACC is lost. If the candidate incorrectly determines that ACC is already lost, then this would be the correct answer.

Distractor: D: is not correct - ACC is lost. If the candidate incorrectly determines that ACC is already lost, then this would still be incorrect since steam cooling with injection would occur when level reached -109 inches, although injection is not reduced until level is restored to above TAF per the transient and accident mitigation guidelines.

Question matches the K/A because the candidate must interpret a reactor water level (which is a low reactor water level) and associate this information to the mechanisms for adequate

core cooling through application of this low reactor water to N1-EOP-2 procedure steps.

REFERENCE:
N1-EOP-9
NER 1M-095

REFERENCES PROVIDED TO THE CANDIDATE:
N1-EOP-2, N1-EOP-9

Question 17 Cross References (table item links)

10CFR55
- 41(b)(10)
- 41(b)(3)

Cognitive Level
- 2

NUREG 1123 KA Catalog Rev. 2
- 295031 EA2.01 4.6*/4.6* Reactor water level

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-006-344-1-02, Rev. NA

OTHER_REFS
- (REPORT) NER-1M-095, Rev. NA

Question Source
- New

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

Question Setting
- C1 (License class closed reference)

18 **SYSID: 21342** **Points: 1.00**

An ATWS is in progress with 37 control rods at position 48. The CRS has directed execution of N1-EOP-3.1, Alternate Control Rod Insertion. Reactor pressure is constant at 1000 psig.

Assuming that each one of the following alternate control rod insertion methods is successful in inserting the control rod(s), which method exerts the GREATEST amount of differential pressure across the drive piston of the stuck control rod drive mechanism(s)?

Control rod insertion is successful:

A. by raising cooling water pressure

B. by venting the over piston volume

C. using the individual rod scram switches

D. using the Reactor Manual Control System

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 18 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 18
System ID:	21342
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-006-344-1-03, EO-1.2, EO-1.3

Answer:	C: is correct - This method would apply the total available differential pressure of the CRD hydraulic system to a single selected control rod. The scram discharge vent and drain valves remain open, the maximum differential pressure is applied over the full travel of the control rod. This would be the greatest d/p as venting the over piston area to atmosphere to a 4" header, which develops less headloss or backpressure than venting through a ¾" hose connection.
Distractor:	A: is not correct - This method increases pressure on the under side of the drive piston but does not vent over piston pressure.
Distractor:	B: is not correct - While this action does significantly raise differential pressure across the drive piston by venting the over-piston area directly to atmosphere, the discharge path is to a ¾" hose connection. The restricted venting path would result in a differential pressure slightly less than discharging to the 4" SDV header.
Distractor:	D: is not correct - This method allows maximizing drive pressure by starting both CRD pumps,

opening the CRD Flow Control Valve, closing the 44-04 Control Rod Drive Water Control Valve (44-04), and closing the Charging Water Header Blocking Valve (44-167). However, these actions do not vent the over piston volume completely.

REFERENCE:
N1-EOP-3.1; Section 7, 6, 5, 3., C-18016-C Sheets 1 and 2

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 18 Cross References (table item links)

10CFR55
- 41(b)(6)

Cognitive Level
- 2

DRW
- C-18016-C Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 295037 EK2.06 3.5/3.6 CRD mechanisms

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-006-344-1-03, Rev. NA

PROC
- N1-EOP-3 Rev. NA
- N1-EOP-3.1 Rev. NA

Question Setting
- C1 (License class closed reference)

19 **SYSID: 21343** **Points: 1.00**

A plant transient occurs while at 100% power, resulting in the following:

- Fuel failure occurs
- Off-site release rates have just reached the Alert level and stabilized
- Secondary containment isolation occurs
- Several Turbine Building Area Radiation Monitors (ARMs) are alarming
- Reactor power is lowered and is now at 75% power

Which one of the following describes the threat to the health and safety of the public and the correct action to protect the health and safety of the public and plant personnel, per the EOPs?

A. Immediate. Scram the reactor and then perform an RPV Blowdown

B. Immediate. Scram the reactor and cooldown within the Tech Spec limit

C. Not immediate. Operate ventilation to prevent a ground level release from the Reactor Building

D. Not immediate. Operate ventilation to maintain the Turbine Building release from an elevated monitored path

Answer: D

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 19 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 19
System ID:	21343
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-006-344-1-06, EO-1.2, EO-1.3

ANSWER: D: is correct - At the ALERT level, the release is sufficiently high that it is not expected to occur during normal plant operation but is sufficiently low such that the condition does not pose an immediate threat to the health and safety of the public.

Distractor: A: is not correct - Not an immediate threat. Scram could be performed but is not required at this time. A RPV Blowdown is not appropriate.

Distractor: B: is not correct - Not an immediate threat. Scram could be performed but is not required at this time.

Distractor: C: is not correct - The concern is within the Turbine Building. Reactor Building negative pressure is being maintained by the RBEVS.

REFERENCE: NER 1M-095; Section 1.9, N1-EOP-6, RR-2

REFERENCES PROVIDED TO THE CANDIDATE: NONE

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 19 Cross References (table item links)

10CFR55

- 41(b)(9)

Cognitive Level

- 2

NUREG 1123 KA Catalog Rev. 2

- 295038 EK1.02 4.2*/4.4* Protection of the general public

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-006-344-1-06 Rev. na

OTHER REFS

- (REPORT) NER-1M-095, Rev. NA

Question Source

- New

PROC

- N1-EOP-6 Rev. NA

Question Setting

- C1 (License class closed reference)

20 **SYSID: 21344** **Points: 1.00**

The plant is at 100% power and has entered N1-SOP-21.1, Fire In Plant. The Station Operator establishes local manual control of FWP #11 FCV.

Per N1-SOP-21.1, which one of the following is the reason for this action?

A. The fire has resulted in as loss of the instrument air system

B. The fire is challenging the safe shutdown capability

C. The fire has affected the ability to reset the HPCI system

D. The fire has burned for 16 minutes and is not under control

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 20 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 20
System ID:	21344
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-006-342-1-01, EO-1.2

Answer: A: is correct - Per N1-SOP-21.1 overrides, if a loss of instrument air occurs is affected, establish manual control of #11/#12 FW FCVs.

Distractor: B: is not correct - Affecting instrument air does not challenge safe shutdown capability. The required action in response to a challenge to safe shutdown capability is scram the reactor.

Distractor: C: is not correct - If the fire has affected HPCI reset, HPCI fuses FU8 and FU9 are pulled to override HPCI. This is not an affect of a degraded instrument air system.

Distractor: D: is not correct - The required action in response to a challenge to safe shutdown capability is scram the reactor.

REFERENCE: N1-SOP-21.1

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 20 Cross References (table item links)

10CFR55

- 41(b)(10)
- 41(b)(7)

Cognitive Level

- 2

NUREG 1123 KA Catalog Rev. 2

- 600000 AK3.04 2.8/3.4; Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE: Actions contained in the abnormal procedure for plant fire on site

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-006-342-1-01 Rev. na

PROC

- N1-SOP-21.1 Rev. NA

Question Setting

- C1 (License class closed reference)

21 **SYSID: 21345** **Points: 1.00**

The following conditions exist:

- Reactor startup is in progress
- Reactor power is on Range 9 of IRMs
- Reactor pressure is 800 psig
- Main Condenser vacuum lowers to 5 inches Hg

Which one of the following is the expected design plant response to the above conditions?

A. MSIVs remain open; MPR controls reactor pressure

B. MSIVs remain open; EPR controls reactor pressure

C. MSIVs close; reactor scram signal from MSIV closure, not high reactor pressure

D. MSIVs close; reactor scram signal from high reactor pressure, not MSIV closure

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 21 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 21
System ID:	21345
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Question Derived from bank question.

O1-OPS-001-255-1-01, EO-1.8b

Answer: C: is correct - A low condenser vacuum will cause a MSIV closure with the mode switch in startup and reactor pressure greater than 600 psig.

Distractor: A: is not correct - MSIVs close.

Distractor: B: is not correct - MSIVs close. MPR is in control.

Distractor: D: is not correct - At 800 psig and range 9 of the IRMs, the only steam flow is through the steam jets and sealing steam. The TBVs are closed. The scram will prevent pressure from reaching the high pressure scram setpoint when the MSIVs close.

REFERENCE: Ops Tech Chapter 10, Chapter 21

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 21 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 1

NUREG 1123 KA Catalog Rev. 2

- 295002 AA2.01 2.9/3.1 Condenser vacuum/absolute pressure

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-255-1-01 Rev. na

Question Source

- Bank

Question Setting

- C1 (License class closed reference)

22 **SYSID: 21346** **Points: 1.00**

Following a loss of feedwater heating, final feedwater temperature lowered by 100°F prior to the operating crew inserting a manual reactor scram.

When performing the post-scram review, which one of the following is the MOST LIMITING thermal limit during the event?

A. LHGR

B. MCPR

C. APLHGR

D. MAPRAT

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 22 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 22
System ID:	21346
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-004-321-1-01, EO-3.0d

Answer: B: is correct - For a loss of feedwater heating, MCPR is the thermal limit of concern.

Distractor: A: is not correct - MCPR is the thermal limit of concern. LHGR is a steady state limit.

Distractor: C: is not correct - MCPR is the thermal limit of concern. APLHGR is a LOCA concern.

Distractor: D: is not correct - MCPR is the thermal limit of concern. MAPRAT relates to APLHGR.

REFERENCE: Ops Tech Chapter 23a, V.B.8
FSAR XV, B.3.2

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 22 Cross References (table item links)

10CFR55

- 41(b)(5)

Cognitive Level

- 1

FSAR

- FSAR Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 295014 AK1.05 3.7/4.2* Fuel thermal limits

Level of Difficulty

- Level 2: System operation and response

LP

- O1-OPS-004-321-1-01, Rev. NA

OTHER_REFS

- OPS TECH, Rev. NA

Question Source

- New

Question Setting

- C1 (License class closed reference)

23 **SYSID: 21347** **Points: 1.00**

Given the following conditions:

- The plant has experienced a failure to scram
- 48 rods remain partially or fully withdrawn
- RPS scram pilot valve power lights are off
- RPV level is stable at 57 inches
- RPV pressure is 920 psig and being controlled by the bypass valves
- Scram air header pressure is currently 0 psig

Which one of the following actions is the correct N1-EOP-3.1, Alternate Control Rod Insertion, action to insert the remaining control rods?

A. Insert repeated manual scram signals

B. Manually initiate Alternate Rod Insertion

C. Pull RPS fuses in cabinet 1S-53 and 1S-55

D. Perform individual rod scrams from the M panel

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 23 Details

Question Type:	*Multiple Choice*
Topic:	RETAKE RO 23
System ID:	21347
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Question Derived from bank question.

O1-OPS-006-344-1-03, EO-1.2

Answer: A: is correct - Manual scram worked partially already, scram air pressure is 0, SDV volume is causing a hydraulic lock, and the scram must be reset to drain the SDV.

Distractor: B: is not correct - This would depressurize the scram air header but will not drain the SDV which is the indicated cause of the failure to scram.

Distractor: C: is not correct - This would de-energize the scram solenoids, but it does not drain the SDV.

Distractor: D: is not correct - Step also de-energizes RPS, which is already done, but it does not drain the SDV.

REFERENCE:
N1-EOP-3.1

REFERENCES PROVIDED TO THE CANDIDATE:
NONE

Question 23 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 2

NUREG 1123 KA Catalog Rev. 2

- 295015 AA1.02 4/4.2* RPS

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-006-344-1-03, Rev. NA

Question Source

- Bank

PROC

- N1-EOP-3 Rev. NA
- N1-EOP-3.1 Rev. NA

Question Setting

- C1 (License class closed reference)

24 **SYSID: 21348** **Points: 1.00**

A plant is starting up following refueling outage with power ascension in progress. With reactor power at 50% the following items occur at the noted times:

09:30 F3-1-2, CONTROL ROD DRIVE PUMP 11 TRIP-VIB
F3-1-5, CRD CHARGING WTR PRESSURE HI/LO

09:31 CRD Pump 12 did not start when its control switch was placed to START

09:37 F3-2-5, CRD ACCUMULATOR LEVEL HIGH PRESS LOW, received for two (2) accumulators.

Per N1-SOP-5.1, Loss of Control Rod Drive, which one of the following is LATEST TIME to insert a manual scram while attempting to restore CRD?

A. 09:31

B. 09:37

C. 09:51

D. 09:57

Answer: D

Associated objective(s):

Development Area (FIO)

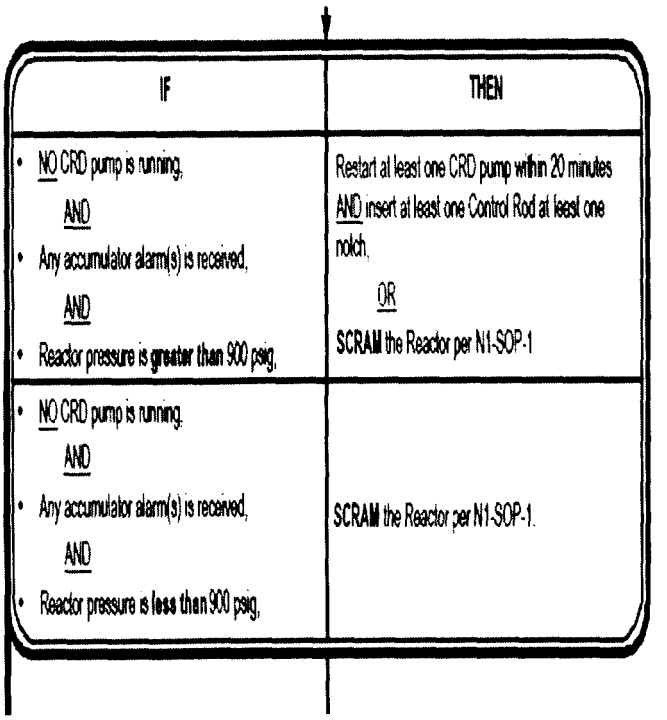
EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 24 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 24
System ID:	21348
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-201-1-01, EO-1.7g

Answer: D. is correct - With reactor pressure above 900 psig (approximately 1000 psig at 50% power) 20 minutes are permitted from receipt of the first accumulator trouble alarm with no CRD pumps running until the scram is required. 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.



IF	THEN
• NO CRD pump is running, AND • Any accumulator alarm(s) is received, AND • Reactor pressure is greater than 900 psig,	Restart at least one CRD pump within 20 minutes AND insert at least one Control Rod at least one notch, OR SCRAM the Reactor per N1-SOP-1
• NO CRD pump is running, AND • Any accumulator alarm(s) is received, AND • Reactor pressure is less than 900 psig,	SCRAM the Reactor per N1-SOP-1.

Distractor: A: is not correct - This answer is plausible if reactor pressure was below 900 psig, but would still be incorrect since the scram below 900 psig reactor pressure is not required until the first accumulator trouble alarm is received. With reactor pressure at 1030 psig, 20 minutes are permitted to establish CRD pressure from receipt of the 1st accumulator trouble alarm.

Distractor: B: is not correct - This is the correct answer with reactor pressure below 900 psig. With reactor pressure at 1030 psig, 20 minutes are permitted to establish CRD pressure from receipt of the 1st accumulator trouble alarm.

Distractor: C: With reactor pressure at 1030 psig, 20 minutes are permitted to establish CRD pressure from receipt of the 1st accumulator trouble alarm not from when it is determined that CRD pumps cannot be started.

cannot be started.

Matches the K/A because the candidate must determine the accumulator trouble alarm is due to low accumulator pressure based on the plant conditions present which include a loss of CRD pumps. Then the candidate must relate this condition to the plant conditions and the procedure requirements outlined in N1-SOP-5.1.

REFERENCE:
N1-SOP-5.1 Flow Chart Override.

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 24 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 2

NUREG 1123 KA Catalog Rev. 2
- 295022 AA2.01 3.5/3.6 Accumulator pressure

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-001-201-1-01 Rev. na

Question Source
- New

PROC
- N1-SOP-5.1 Rev. NA

Question Setting
- C1 (License class closed reference)

25 **SYSID: 21349** **Points: 1.00**

The plant is at power when the following occur:

- A leak occurs in the Reactor Water Cleanup System bringing in K3-3-4, CLEAN-UP SYSTEM SYSTEM LEAK AREA T HI.
- Cleanup 33-01, CU RETURN ISOLATION VALVE 1 (INSIDE), closed
- Cleanup 33-02, CU SUPPLY ISOLATION VALVE 11 (INSIDE), won't close
- Cleanup 33-04, CU SUPPLY ISOLATION VALVE 12 (OUTSIDE), won't close
- Area temperatures in RB 261' east are 145°F
- Area temperatures in RB 281' east are 137°F

Based on the above conditions, which one of the following is the correct action at this time?

A. Scram the reactor, and then cool down at a rate less than 100°F/hour

B. Scram the reactor, and then immediately enter EOP-8, RPV Blowdown

C. Initiate a normal plant shutdown per OP-43C, and be less than 212°F in 10 hours

D. Continue efforts to close Cleanup 33-04, and perform an emergency power reduction

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 25 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 25
System ID:	21349
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-006-344-1-05, EO-1.2

Answer: B. is correct - Cleanup is not isolated; a primary system discharging into the reactor building. 33-01 and a check valve are the isolations for the return to the RPV and 33-02 and 33-04 are the isolations for the supply from the RPV. A scram and then blowdown are required because two general areas are above Max Safe temperature of 135°F.

Distractor: A: is not correct - RPV Blowdown is required since a second area is above Max Safe for the same parameter. I.e., two area temperatures or two area radiation levels above max safe not one area temperature and one area radiation level above max safe).

Distractor: C: is not correct - If is it incorrectly determined that Cleanup is isolated, then this path in EOP-5 includes this action.

Distractor: D: is not correct - The cleanup valve closure can be misinterpreted as an isolated system because of the closed valve and incorrectly determining the location of the check valve (isolation). Also, it can be misinterpreted that the area temperatures are below the max safe values by using the wrong Detail

"T" versus Detail "S".

REFERENCE: N1-EOP-5

REFERENCES PROVIDED TO THE CANDIDATE: N1-EOP-5

MODIFIED: Modified the question so two area temperatures are above the max safe values requiring a scram and RPV Blowdown, and not just a scram. Change distracter "D" from a required TS shutdown based on Reactor Coolant Isolation Valves, to "Continue efforts to close Cleanup 33-04. A shutdown or scram *are not required*" since the cleanup valve closures can be misinterpreted as an isolated system because of the closed valve and *incorrectly determining* the location of the check valve (isolation).

Question 25 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 3

NUREG 1123 KA Catalog Rev. 2
- G2.4.6 3.1/4 Knowledge symptom based EOP mitigation strategies
- 295032 High Secondary Containment Area Temperature

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-006-344-1-05 Rev. na

Question Source
- Modified

PROC
- N1-EOP-5 Rev. NA

Question Setting
- C1 (License class closed reference)

26 **SYSID: 21350** **Points: 1.00**

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

- Rx Bldg Supply Fan 11 *running in* FAST
- Rx Bldg Exhaust Fan 11 running in FAST
- Rx Bldg Supply Fan 12 in STOP
- Rx Bldg Exhaust Fan 12 in STOP
- The fan drive belt on Rx Bldg Exhaust Fan 11 breaks

Which one of the following is the effect on Secondary Containment?

A. Secondary Containment is lost until operator action is taken

B. Secondary Containment is inoperable until the fan belt is replaced

C. Rx Bldg Exhaust Fan 12 auto starts to maintain Secondary Containment

D. Rx Bldg Supply Fan 11 suction damper auto closes to maintain Secondary Containment

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 26 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 26
System ID:	21350
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-288-1-01, EO-1.8b

Answer:	A: is correct - A loss of the operating reactor building exhaust fan would not trip the supply fan, the operating supply fan would raise secondary containment pressure above -.25 inches of water and approach a positive pressure. Secondary Containment integrity is lost. The operator can either start Exhaust Fan #12 or either RBEVS train to restore and maintain Rx Bldg negative d/p.
Distractor:	B: is not correct - Reactor Building fans are not required to be operable by Tech Specs for Secondary Containment operability. As soon as operator action is taken to either start the other exhaust fan or one of the RBEVS trains, Secondary Containment integrity is restored. Fixing the fan belt is not required to restore Secondary Containment integrity.
Distractor:	C: is not correct - The standby fan does not start on low reactor building differential
Distractor:	D: is not correct - The supply fan suction dampers cannot reposition to make the reactor building pressure negative.

REFERENCE: N1-OP-10, Ops Tech Chapter 13a, N1-ARP-L1, L1-1-5, L1-3-4

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 26 Cross References (table item links)

10CFR55
- 41(b)(9)

Cognitive Level
- 1

NUREG 1123 KA Catalog Rev. 2
- 295035 EK1.01 3.9/4.2* Secondary containment integrity

Level of Difficulty
- Level 2: System operation and response

LP
- O1-OPS-001-288-1-01 Rev. na

OTHER_REFS
- OPS TECH, Rev. NA

Question Source
- New

PROC
- N1-ARP-L1 Rev. NA
- N1-OP-10 Rev. NA

Question Setting
- C1 (License class closed reference)

27 **SYSID: 21351** **Points: 1.00**

The plant has experienced a LOCA. The following conditions currently exist in the Primary Containment:

- Drywell Pressure 3.1 psig
- Drywell Temperature 230°F
- Drywell H2 Concentration 5.0%
- Torus H2 Concentration 7.0%
- Drywell O2 concentration 3.2%
- Torus O2 Concentration 4.2%
- Torus Pressure 2.5 psig
- Torus Level 12.5 feet

Offsite radioactivity release rate will remain below release limits. Which one of the following is the correct action if the Torus cannot be vented using the desired vent path?

A. Vent the Torus through the RBEVS

B. Vent the Drywell using the Vent and Purge Fan

C. Purge the Torus with nitrogen at maximum flow

D. Purge the Drywell with nitrogen at maximum flow

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 27 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 27
System ID:	21351
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-006-344-1-19, EO-1.2

Answer: B. is correct - The candidate must recognize H2 parameters are over their limits and then recognize the torus as the preferred path for venting, per EOP enter steps 31 and 34, with torus level <27 ft and torus pressure < 3.0 psig, vent the torus through the vent and purge fan. If the torus cannot be vented, then vent the drywell through the vent and purge fan. Purging of containment is not established until it is known that the containment can be vented.

Distractor: A: is not correct - This is permissible if drywell pressure were 0.2 psig lower; less than 3 psig.

Distractor: C: is not correct - Action per Block 34 after venting is successful. Purging of containment is not established until it is known that the containment can be vented.

Distractor: D: is not correct - Action per Block 31 after venting is successful. Purging of containment is not established until it is known that the containment can be vented.

REFERENCE: N1-EOP-4.2

REFERENCES PROVIDED TO THE CANDIDATE:
N1-EOP-4.2

Question 27 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 3

NUREG 1123 KA Catalog Rev. 2

- 500000 EK2.07 3.2/3.7 Drywell vent system

Level of Difficulty

- Level 4: Highest order knowledge

LP

- O1-OPS-006-344-1-19, Rev. NA

Question Source

- New

PROC

- N1-EOP-4.2 Rev. NA

Question Setting

- C1 (License class closed reference)

28 **SYSID: 21352** **Points: 1.00**

During an outage:

- SDC system 11 in service
- Safety Assessment Level for DHR is currently N+1
- Recirc Pumps #14 and #15 are operating
- The RPV vessel head was just installed
- SDC Pump #11 trips and NO SDC system can be started

Per N1-SOP-6.1, Loss of SFP/Rx Cavity Level/Decay Heat Removal, which one of the following is the correct initial and contingency action in response to the loss of SDC?

A. Close 70-49, BV-11 SDC HX RBCLC INLET, and raise RPV water level to at least the elevation of the Main Steam Line Drains

B. Open 70-85, BV-CU NONREGEN HX RBCLC INLET, and if necessary establish a feed and bleed using the CRD and Cleanup systems

C. Close 70-49, BV-11 SDC HX RBCLC INLET, and if the time to boil estimation is less than 8 hours then start another reactor recirc pump

D. Open 70-85, BV-CU NONREGEN HX RBCLC INLET, and if necessary establish a feed and bleed using Containment Spray and vessel drains to the Drywell Floor Drain Tank

Answer: B

Associated objective(s):

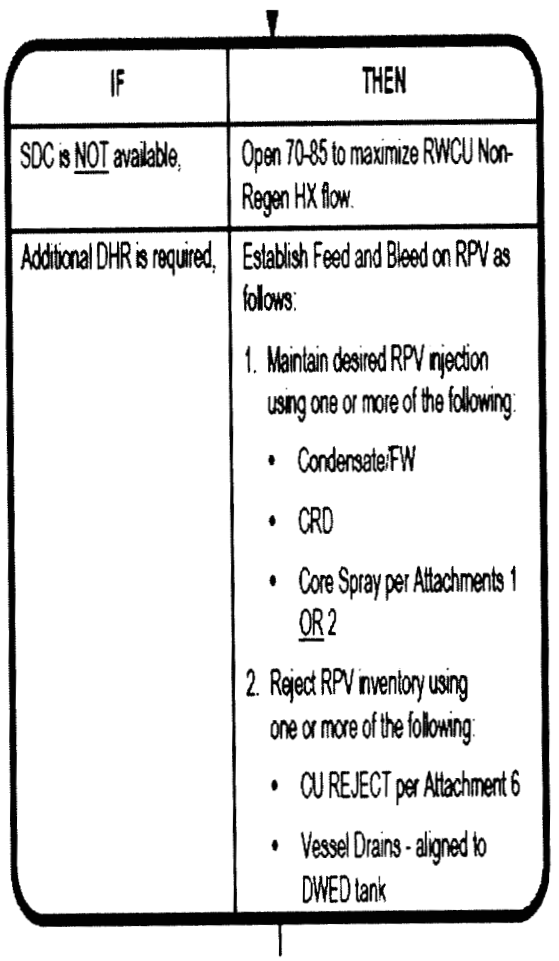
Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 28 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 28
System ID:	21352
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Question Derived from bank question.
	O1-OPS-001-205-1-01, EO-1.7g, EO-1.8a



IF	THEN
SDC is NOT available,	Open 70-85 to maximize RWCU Non-Regen HX flow.
Additional DHR is required,	Establish Feed and Bleed on RPV as follows: 1. Maintain desired RPV injection using one or more of the following: • Condensate/FW • CRD • Core Spray per Attachments 1 OR 2 2. Reject RPV inventory using one or more of the following: • CU REJECT per Attachment 6 • Vessel Drains - aligned to DWED tank

Answer: B: is correct

Distractor: A: is not correct The Cleanup BV is opened. Reactor water level is only required to be raised if NO recirc pumps are operating.

Distractor: C: is not correct - The Cleanup BV is opened. For a loss of SDC, starting a third reactor recirc pump is prohibited.

Distractor: D: is not correct - Containment Spray is not a system that is used for feed and bleed. Can use Condensate, FW, CRD, or Core Spray and bleed through the vessel drains to the Drywell Equipment Drain Tank, not the Drywell Floor Drain Tank.

REFERENCE: N1-SOP-6.1

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 28 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 1

NUREG 1123 KA Catalog Rev. 2
- 205000 A2.06 3.4/3.5 SDC/RHR pump trips

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-001-205-1-01 Rev. na

Question Source
- Bank

PROC
- N1-SOP-6.1, Rev. NA

Question Setting
- C1 (License class closed reference)

29 **SYSID: 21353** **Points: 1.00**

The plant is at 100% power with FWP 12 in STANDBY. The following events occur:

At time = 0	FWP 11 Trips
At time = 10 seconds	Reactor scrams on low level
At time = 15 seconds	Main Turbine trips

Which one of the following is the correct signal that starts FWP 12 and the correct value at which RPV level will be controlled following the scram as FWP 13 coasts down?

FWP 12 will start when the:

A. low level scram signal is received and controls RPV level at 72 inches

B. low level scram signal is received and controls RPV level at 65 inches

C. main turbine trip signal is received and controls RPV level at 72 inches

D. main turbine trip signal is received and controls RPV level at 65 inches

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 29 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 29
System ID:	21353
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-259-1-01, EO-1.4c, EO-1.5b

Answer: A. is correct - The reactor scram signal on low RPV level (53") signals the motor-driven FW pump to start. This is normally a backup signal to the turbine trip signal but occurs first for the conditions provided. Level control for FWP 12 is automatically set to +72" during the HPCI mode of operation.

Distractor: B: is not correct - FWP 12 will control at 72 inches in the HPCI mode. FWP 11 controls at 65 inches in the HPCI mode.

Distractor: C: is not correct - The HPCI start signal occurs on low level which occurred prior to the main turbine trip signal at T = 15 seconds.

Distractor: D: is not correct - The HPCI start signal occurs on low level which occurred prior to the main turbine trip signal at T = 15 seconds. FWP 12 will control at 72 inches in the HPCI mode. FWP 11 controls at 65 inches in the HPCI mode.

Question matches the K/A because the candidate must know the design features of HPCI (which pump controls at which level) and the interlocks (signals that initiate HPCI) to answer the question.

REFERENCE: N1-OP-16, Section B.2.0

MODIFIED: The original question had FWP 12 running and FWP 11 in standby, then asked the FWP 11 start signal and control band. The correct answer was "b". The question was modified to have FWP 11 running and FWP 12 in standby, and asks the FWP 12 start signal and control band. The correct answer is "a".

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 29 Cross References (table item links)

10CFR55
- 41(b)(7)

Cognitive Level
- 1

NUREG 1123 KA Catalog Rev. 2
- 206000 K4.07 4.3*/4.3 Automatic system initiation: BWR-2,3,4

Level of Difficulty
- Level 2: System operation and response

LP
- O1-OPS-001-259-1-01 Rev. na

Question Source
- Bank

PROC
- N1-OP-16 Rev. NA

Question Setting
- C1 (License class closed reference)

30 **SYSID: 21354** **Points: 1.00**

Which one of the following set of conditions requires entry into Tech Spec 3.1.3, Emergency Cooling System?

A. One of the Emergency Condenser Radiation Monitors (EC Vent Rad Monitor 111) is inoperable

B. Performance of N1-ST-Q4, Reactor Coolant System Isolation Valves Operability Test

C. Performance of N1-ST-M2, Emergency Cooling System Makeup Tank Level Control Valves Exercising Test

D. During RPV cool down using Emergency Cooling, the RPV cool down rate exceeds 100°F in a one hour period

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 30 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 30
System ID:	21354
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-207-1-01, EO-1.7f

Answer: B: is correct - Per N1-ST-Q4, section 8.1, EC Loop 11 IVs Operability Test 8.1.0: Notify SSS to enter T.S. 3.1.3b which is the most restrictive LCO for the valves tested in this section.
Per N1-ST-Q4, section 8.2, EC Loop 12 IVs Operability Test 8.2.0: Notify SSS to enter T.S. 3.1.3b which is the most restrictive LCO for the valves tested in this section.
Per N1-ST-Q4, section 8.4, EC Makeup LCV Operability Test 8.4.1:Isolating the EC level control valve from the EC renders the associated EC Loop inoperable T.S. 3.1.3.b applies.

Distractor: A: is not correct - OP-13, D.9.0: Whenever the EC System is required to be operable; the Noble Gas Activity Monitors (EC Vent Rad Monitor) shall also be operable (ODCM Table D 3.6.14-2). There are 4 EC Vent Rad Monitors with 2 for each EC (111, 121, 112, and 122). One monitor inoperable does not require declaring the affected EC inoperable.

Distractor: C: is not correct - During the performance of subsections 8.1 and 8.2 of the ST, the emergency condenser level control valves will be cycled causing the emergency condenser level to increase slightly, but there is no TS 3.1.3 implication.

Distractor: D: is not correct - TS 3.2.1, Reactor

Vessel Heatup and Cooldown Rates, is entered, not TS 3.1.3.

REFERENCE: N1-ST-Q4, Section 8.1, 8.2, 8.4

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 30 Cross References (table item links)

10CFR55

- 41(b)(8)

Cognitive Level

- 2

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
- 207000 Isolation (Emergency) Condenser

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-207-1-01 Rev. NA

Question Source

- New

PROC

- N1-ST-Q4 Rev. NA

Question Setting

- C1 (License class closed reference)

31 **SYSID: 21355** **Points: 1.00**

The plant is at 100% power when the following annunciator alarms:

- K1-1-2, EMER COND VENT 11 RAD MONITOR
- ATC RO confirms EMERG COND RMON 111 and 112 are at 10 mrem/hr

Which one of the following indications confirms an EC tube leak?

A. Shell side temperature rising, shell side level rising

B. Shell side temperature rising, shell side level lowering

C. Shell side temperature lowering, shell side level rising

D. Shell side temperature lowering, shell side level lowering

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 31 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 31
System ID:	21355
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-207-1-01, EO-1.7e, EO-1.8b

Answer: A: is correct - EC tube leak is from the primary side (tube side) which is at reactor pressure to the secondary side (shell side) which is at atmospheric pressure. With a tube leak, shell side temperature rises and shell side level rises.

Distractor: B: is not correct - Shell side level rises, not lowers. Lowering shell side level is an indication of a leak on the secondary side of the EC, but not a tube leak.

Distractor: C: is not correct - Shell side temperature rises, not lowers.

Distractor: D: is not correct - Shell side temperature rises, not lowers. Shell side level rises, not lowers. Lowering shell side level is an indication of a leak on the secondary side of the EC, but not a tube leak.

Question matches the K/A because of the two temperatures indicated, primary and shell side temperatures, the shell side temperature is of higher importance for the conditions specified in the question. Therefore, the "and" element is addressed by evaluating the more important element of the two elements. Furthermore, the candidate must demonstrate the ability to monitor the shell side **temperature** through an understanding of how it changes to confirm the EC tube leak. Additionally, an understanding of the shell side **level** change is also

required to confirm the EC tube leak.

REFERENCE: N1-ARP-K1 K1-3-3, K1-3-2, K1-2-5

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 31 Cross References (table item links)

10CFR55
- 41(b)(8)

Cognitive Level
- 2

NUREG 1123 KA Catalog Rev. 2
- 207000 A4.03 3/3.2 Primary and shell side temperatures: BWR-2,3

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-001-207-1-01 Rev. NA

Question Source
- New

PROC
- N1-ARP-K1 Rev. NA

Question Setting
- C1 (License class closed reference)

32 **SYSID: 21356** **Points: 1.00**

During the performance of the Quarterly Core Spray Pump and Valve Operability Tests, the following are performed:

- 40-12, Core Spray Discharge IV 11 (Outside) is verified open
- Breaker for 40-30 is unlocked and positioned to ON
- 40-30, Core Spray Vent Isolation Valve 111, is opened
- 40-32, Core Spray Vent Isolation Valve 112, is opened
- After flow is verified 40-32, is closed
- 40-30, is closed
- Breaker for 40-30 is locked OFF

Which one of the following is the location an operator is stationed at and the reason the operator is stationed at this location when performing the above actions?

An operator is stationed at:

A. 40-26, 11 Keep Fill Vent Blocking Valve, to fully close it if a Loss of Offsite Power occurs

B. 40-26, 11 Keep Fill Vent Blocking Valve, to fully close it if an Appendix R event occurs

C. 40-30, Core Spray Vent Isolation Valve 111, breaker to place it to off if a Loss of Offsite Power occurs

D. 40-30, Core Spray Vent Isolation Valve 111, breaker to place it to off if an Appendix R event occurs

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 32 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 32
System ID:	21356
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-209-1-01, EO-1.7f

Answer: B: is correct - Per N1-ST-Q1A/B/C/D, cautions: To satisfy Appendix R requirements, an operator with a VA-1 Key shall be stationed at 40-26, 11 KEEP FILL VENT BLOCKING VALVE, to quickly close the valve to mitigate an Appendix R event if directed by the Control Room. The operator shall remain stationed until IV 40-30 is closed AND the breaker for IV 40-30 is locked in the OFF position.

Reference SE 98-021, Draft B Revision 0, Appendix R High/Low Pressure and Inventory Loss Flow Path Isolation for the Core Spray System Injection High Point Vent Lines

Distractor: A: is not correct - 40-26 is normally throttled and must be manually closed if an Appendix R event occurs. There are precautions and limitations for Core Spray when a loss of offsite power occurs; however, they are related to EDG normal and emergency loading.

Distractor: C: is not correct - 40-26 is normally throttled and must be manually closed if an Appendix R event occurs. There are precautions and limitations for Core Spray when a loss of offsite power occurs; however, they are related to EDG normal and emergency loading. The operator is stationed at 40-26 and not at the breaker for 40-30.

Distractor: D: is not correct - 40-26 is normally throttled and must be manually closed if an Appendix R event occurs. The operator is stationed at 40-26 and not at the breaker for 40-30.

REFERENCE: N1-ST-Q1A/B/C/D

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 32 Cross References (table item links)

10CFR55

- 41(b)(10)
- 41(b)(7)

Cognitive Level

- 1

NUREG 1123 KA Catalog Rev. 2

- G2.1.32 3.4/3.8 Ability to explain and apply system limits and precautions
- 209001 Low Pressure Core Spray System

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-209-1-01 Rev. NA

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: 21357** **Points: 1.00**

A loss of coolant accident has occurred with a loss of PB103. Current plant conditions:

- Containment Sprays are in service
- RPV Blowdown has been performed
- Drywell pressure is 7.5 psig
- Torus pressure is 6.5 psig
- Torus water level is 11.2 feet
- Torus water temperature is 200°F

Which one of the following is the approximate flow limit for each core spray loop for the above conditions?

A. 175×10^4 lbm/hr.

B. 240×10^4 lbm/hr.

C. 350×10^4 lbm/hr.

D. 480×10^4 lbm/hr.

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 33 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 33
System ID:	21357
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-006-344-1-02, EO-1.3

Answer: A. is correct - For the initial plant conditions:
Torus Overpressure = Torus Press + 0.433(Torus Level – 4.5)
Torus Overpressure = 6.5 + 0.433(11.2 - 4.5)
Torus Overpressure = 6.5 + 0.433(6.7)
Torus Overpressure = 6.5 + 2.9011 = 9.4011 psig
So, the 5 psig curve is used on Detail N1. With one subsystem available per loop, the max flow is approximately 175 x 10^4 lbm/hr for each subsystem.

Distractor: B: is not correct - This is the limit if the 10 psig rather than the 5 psig curve is incorrectly used and the one subsystem limit which is the correct limit is used.

Distractor: C: is not correct - This is the limit using the 5 psig curve but incorrectly determining two subsystems rather than only one subsystem.

Distractor: D: is not correct - This is the limit if the 10 psig rather than the 5 psig curve is incorrectly used and the two subsystem limit is also incorrectly used.

REFERENCE:
N1-EOP-2, Figure N1.
NER-1M-095 page 49 and 50

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

REFERENCES PROVIDED TO THE CANDIDATE:
N1-EOP-2

Question 33 Cross References (table item links)

10CFR55
- 41(b)(8)

Cognitive Level
- 3

NUREG 1123 KA Catalog Rev. 2
- 209001 K1.02 3.4/3.4 Torus/suppression pool

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-006-344-1-02, Rev. NA

OTHER_REFS
- (REPORT) NER-1M-095, Rev. NA

Question Source
- New

PROC
- N1-EOP-02 Rev. NA

Question Setting
- C1 (License class closed reference)

34 **SYSID: 21358** **Points: 1.00**

During an ATWS, the CRS directs boron injection be initiated. The key lock switch for Liquid Poison (LP) is turned to the START PUMP 11 position.

- LP Pump 11 starts
- LP 11 explosive valve fires
- LP 12 explosive valve is not fired
- Cleanup inboard isolation valve (33-02) closes
- Cleanup outboard isolation valve (33-04) remains open

Which one of the following is the correct evaluation of the above conditions?

A. Both the LP system and the Cleanup system responded per design

B. Both the LP system and the Cleanup system did not respond as expected

C. The LP system responded per design. The Cleanup system did not respond as expected

D. The Cleanup systems responded per design. The LP system did not respond as expected

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 34 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 34
System ID:	21358
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-211-1-01, EO-1.4c, EO-1.8b

Answer: B: is correct - Although only the 11 LP pump starts, auxiliary relays send a signal for both explosive valves to fire. A full isolation signal is sent to the Cleanup system.

Distractor: A: is not correct - Neither system responded as expected. Cleanup 33-04 should have closed and LP 12 explosive valve should have fired.

Distractor: C: is not correct - Neither system responded as expected. Cleanup 33-04 should have closed and LP 12 explosive valve should have fired.

Distractor: D: is not correct - Neither system responded as expected. Cleanup 33-04 should have closed and LP 12 explosive valve should have fired.

REFERENCE: N1-OP-12, H.1.3, and Section B

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 34 Cross References (table item links)

10CFR55
- 41(b)(6)

Cognitive Level
- 1

NUREG 1123 KA Catalog Rev. 2
- 211000 A3.08 4.2*/4.2* System initiation: Plant-Specific

Level of Difficulty
- Level 2: System operation and response

LP
- O1-OPS-001-211-1-01 Rev. NA

Question Source
- New

PROC
- N1-OP-12 Rev. NA

Question Setting
- C1 (License class closed reference)

35 **SYSID: 21359** **Points: 1.00**

With the plant operating at 60% reactor power, the CRS directs a manual reactor scram be inserted. Which one of the following describes how RPS functions to complete the control rod insertion when the Reactor Mode Switch is placed to SHUTDOWN?

A. ONLY the manual scram channels are actuated. A 13.4-second time delay prevents resetting the scram too soon

B. ONLY the automatic scram channels are actuated. A 25-second time delay prevents resetting the scram too soon

C. BOTH the manual and automatic scram channels are actuated. A 13.4-second time delay prevents resetting the scram too soon

D. BOTH the manual and automatic scram channels are actuated. A 25-second time delay prevents resetting the scram too soon

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 35 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 35
System ID:	21359
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-212-1-01, *EO-1.4b, EO-1.4c*

Answer: A: is correct -Reactor Mode Switch contacts are only in the manual scram circuits. There are two time delays related to RPS, one for the scram channels (13.4 seconds) and one for ATWS/ARI actuation (25 seconds). Only the 13.4 second time delay is actuated.

Distractor: B: is not correct - Reactor Mode Switch contacts are only in the manual scram circuits. There are two time delays related to RPS, one for the scram channels (13.4 seconds) and one for ATWS/ARI actuation (25 seconds). Only the 13.4 second time delay is actuated.

Distractor: C: is not correct - Reactor Mode Switch contacts are only in the manual scram circuits. Automatic scram circuits are not actuated.

Distractor: D: is not correct - Reactor Mode Switch contacts are only in the manual scram circuits. Automatic scram circuits are not actuated. There are two time delays related to RPS, one for the scram channels (13.4 seconds) and one for ATWS/ARI actuation (25 seconds). Only the 13.4 second time delay is actuated.

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

REFERENCE: RPS prints, N1-OP-40
C-19859-C, Elementary Wiring Diagram Reactor Protection System

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 35 Cross References (table item links)

10CFR55
- 41(b)(7)

Cognitive Level
- 2

DRW
- C-19859-C Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 212000 K4.08 4.2*/4.2* Complete control rod insertion following SCRAM signal generation

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-001-212-1-01 Rev. na

Question Source
- New

PROC
- N1-OP-40 Rev. NA

Question Setting
- C1 (License class closed reference)

36 **SYSID: 21360** **Points: 1.00**

A plant startup and heatup are in progress, with the following:

- IRM channel 11 is reading 86/125 on range 7
- REFUEL INST TRIP BYPASS 11 switch in COINCIDENT
- REFUEL INST TRIP BYPASS 12 switch in COINCIDENT
- The operator down ranges IRM 11 to range 6

Which one of the following identifies the designed plant response?

A. Only an IRM alarm

B. Only a rod block

C. Rod block and only a half scram

D. Rod block and full reactor scram

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 36 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 36
System ID:	21360
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-212-1-01, EO-1.4c, EO-1.5b

Answer: C: is correct - Upscale neutron flux level Scram. The trips of four of the IRM monitors are incorporated in logic channel 11 and the trips of the other four IRM monitors are incorporated in logic channel 12. With the REFUEL INST TRIP BYPASS 11 and 12 switches in the NON-COINCIDENT position, a single IRM channel INOP or UPSCALE TRIP will cause a Reactor Scram. In the COINCIDENT position, both channel 11 and channel 12 RPS must trip. When the IRM is down ranged, it is upscale on the lower range.

Distractor: D: is not correct - If in non-coincident a full scram would have occurred.

Distractor: A: is not correct - The lower scale (range 6) is upscale based on the range 7 reading; alarm, rod block, half scram.

Distractor: B: is not correct - The lower scale (range 6) is upscale based on the range 7 reading; alarm, rod block, half scram.

REFERENCE: C-19859-C, N1-OP-38A

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 36 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 2

DRW

- C-19859-C Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 215003 A4.07 3.6/3.6 Verification of proper functioning/ operability

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-212-1-01 Rev. na

Question Source

- New

PROC

- N1-OP-38A Rev. NA

Question Setting

- C1 (License class closed reference)

37 **SYSID: 21361** **Points: 1.00**

A plant startup is in progress, with the following:

- All SRMs are reading about 700 cps
- SRM detectors are fully inserted
- IRMs are on Range 1
- THEN, a loss of power to 24 VDC Bus 11 occurs

Which one of the following identifies the effect of the power loss on SRMs?

A. SRM 11 and 12 are de-energized preventing trips from these SRMs

B. SRM 11 and 12 are de-energized causing trips from these SRMs

C. SRM 13 and 14 are de-energized preventing trips from these SRMs

D. SRM 13 and 14 are de-energized causing trips from these SRMs

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 37 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 37
System ID:	21361
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-215-1-02, EO-1.4

Answer: B: is correct - SRM 11 and 12 are powered from Bus 11. The loss of power results in generation of SRM trip functions such as a rod block not the prevention of trips.

Distractor: A: is not correct - On loss of power the SRM trip auxiliary units generate trip signals.

Distractor: C: and D: are not correct - SRMs 13 and 14 are powered from 24VDC Bus 12, not Bus 11. These SRMs are not affected by the loss of Bus 11.

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 37 Cross References (table item links)

10CFR55
- 41(b)(2)
- 41(b)(7)

Cognitive Level
- 2

NUREG 1123 KA Catalog Rev. 2
- 215004 K2.01 2.6/2.8 SRM channels/detectors

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-001-215-1-02 Rev. NA

Question Source
- New

PROC
- N1-OP-47B Rev. NA

Question Setting
- C1 (License class closed reference)

38 **SYSID: 21362** **Points: 1.00**

The plant is at 100% power. Following APRM gain adjustments per N1-REP-12, APRM Gain Adjustment, APRM Gain Adjustment Factors are as follows:

- APRM 11 = 0.99
- ALL other APRMs = 1.00
- Heat Balance = 1850 Mwth

Which one of the following describes (1) the approximate APRM 11 indication at the E Panel (% power) and, (2) whether the margin to the flow-biased scram set point for APRM 11 is GREATER THAN or LESS THAN the other APRM channels?

A. (1) 99%
(2) LESS THAN

B. (1) 99%
(2) GREATER THAN

C. (1) 101%
(2) LESS THAN

D. (1) 101%
(2) GREATER THAN

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 38 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 38
System ID:	21362
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-215-1-02, EO-1.11a

Answer: C. is correct -AGAF is the ratio of actual thermal power over APRM indicated power. (100% power) / (0.99 GAF) = ~101% indicated power. An APRM reading less than actual power (AGAF>1.0) would be further from its flow biased scram setpoint and therefore non-conservative. An APRM reading greater than actual power (AGAF<1.0) would be closer to its flow biased scram setpoint and therefore conservative. APRM 11 indicated power is 101% which is above the other APRM values (100%). It is also 1% closer to the flow biased scram set point (which is calculated but the same for all APRMs based upon flow), and therefore the margin to the flow-biased scram set point is LESS THAN the other APRMs.

Distractor: A: is not correct - AGAF is the ratio of actual thermal power over APRM indicated power. (100% power) / (0.99 GAF) = ~101% indicated power.

Distractor: B: is not correct - AGAF is the ratio of actual thermal power over APRM indicated power. (100% power) / (0.99 GAF) = ~101% indicated power. An APRM reading greater than actual power (AGAF<1.0) would be closer to its flow biased scram setpoint.

Distractor: D: is not correct - An APRM reading greater than actual power (AGAF<1.0) would be closer to its flow biased scram set point.

Question matches K/A because the candidate must monitor APRM changes following a GAF adjustment and make predictions based on the information resulting from the adjustment. The candidate must predict the expected APRM readings and then determine the margin to the flow-biased scram set point for APRM 11 as compared to the other APRMs.

REFERENCE:
N1-REP-12

REFERENCES PROVIDED TO THE CANDIDATE:
NONE

Question 38 Cross References (table item links)

10CFR55
- 41(b)(7)

Cognitive Level
- 2

NUREG 1123 KA Catalog Rev. 2
- 215005 A1.07 3/3.4 APRM (gain adjustment factor)

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-001-215-1-02 Rev. NA

Question Source
- New

PROC
- N1-REP-12 Rev. NA

Question Setting
- C1 (License class closed reference)

39 **SYSID: 21363** **Points: 1.00**

Given the following conditions:

- Reactor power is 70% operating on the 100% Rod Line
- Recirc Master controller is controlling all five (5) Reactor Recirc Pump speeds
- APRM Flow Unit 11 output lowers to 50% and remains steady at 50%
- Reactor Recirc Pump speeds are unchanged

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

A. Rod block. No ½ scram

B. Rod block and ½ scram

C. Only a flow unit alarm at the F Panel

D. Only a flow unit alarm at the G Panel

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 39 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 39
System ID:	21363
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-215-1-02, EO-1.5b, EO-1.8b

Answer: A: is correct - Flow-biased rod block of 1.287W + 4.472% and flow-biased scram set point of 1.287W+ 13.54% are in effect. A lowering flow signal causes W = 50; and results in setpoints of 64.35+4.472%=68.822% (ROD BLOCK) and 64.35+13.54%=77.89% (SCRAM).

Distractor: B: is not correct - Above the rod block set point but below the scram set point.

Distractor: C: is not correct - Rod block is received because the flow-biased rod block set point is exceeded.

Distractor: D: is not correct - Degraded condition would be alarmed at the F panel. Rod block is received because the flow-biased rod block set point is exceeded. Also, greater than 6.8% mismatch causes rod block.

REFERENCE: N1-ARP-F2-2-6; N1-OP-38C, Section B, TS Flow-Biased set points

REFERENCES PROVIDED TO THE CANDIDATE:
N/F MAP (5 Loop) F-45683-C

Question 39 Cross References (table item links)

10CFR55

- 41(b)(10)
- 41(b)(7)

Cognitive Level

- 2

DRW

- F-45683-C, Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 215005 K6.07 3.2/3.3 Flow converter/comparator network: Plant-Specific

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-215-1-02 Rev. NA

Question Source

- New

PROC

- N1-ARP-F2 Rev. NA
- N1-OP-38C Rev. NA

Question Setting

- C1 (License class closed reference)

40 **SYSID: 21364** **Points: 1.00**

With the plant operating at 100% power

- I&C reports the 2-1 relay in the 11-1 ADS logic circuit is burned-out and will not perform its intended safety function
- No other ADS circuitry problems exist

Assuming an ADS initiation signal is received and remains valid (sustained), which one of the following is the correct effect of this relay failure on the ADS response?

A. The primary valves open when the initiation signal is present for 115.5 seconds and no sooner. The secondary valves do not open.

B. The secondary valves open when the initiation signal is present for 115.5 seconds and no sooner. The primary valves do not open.

C. The primary valves open as soon as the initiation signal is present for 111 seconds. The secondary valves do not open.

D. The secondary valves open as soon as the initiation signal is present for 111 seconds. The primary valves do not open.

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 40 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 40
System ID:	21364
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Question Derived from bank question.

O1-OPS-001-218-1-01, EO-1.4b, EO-1.8b

Answer: C: is correct - The primary valves open as soon as the initiation signal is present for 111 seconds, not 115.5 seconds. The secondary valves do not open.

Distractr: A: is not correct - The primary valves will open when the initiation signal has been present for 111 seconds, rather than 115.5 seconds.

Distractor: B: is not correct - The secondary valves will not open.

Distractor: D: is not correct - The secondary valves will not open.

REFERENCE: DWG C-19859-C, Sheets 18, 18A, 24 and 24A, P&ID C-18015-C

REFERENCES PROVIDED TO THE CANDIDATE:
DWG C-19859-C, Sheets 18, 18A, 24 and 24A

Question 40 Cross References (table item links)

10CFR55
- 41(b)(7)

Cognitive Level
- 3

DRW
- C-19859-C Rev. NA
- P&ID C-18015-C Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 218000 A3.01 4.2*/4.3 ADS valve operation

Level of Difficulty
- Level 4: Highest order knowledge

LP
- O1-OPS-001-218-1-01 Rev. NA

Question Source
- Bank

Question Setting
- C1 (License class closed reference)

41 **SYSID: 21365** **Points: 1.00**

Given the following plant conditions:

- A Core spray inside IV packing leak has caused drywell pressure to rise to 4 psig
- The reactor has scrammed due to high Drywell pressure
- When performing N1-SOP-40.2, Vessel/Containment Isolation, you observe NO LIGHT INDICATION for valves 40-30, Core Spray Vent Isolation Valve 111, and 40-31, Core Spray Vent Isolation Valve 121

Which one of the following is the correct action in response to this absence on indicating lights on 40-30 and 40-31?

A. An operator must be sent to manually close 40-30 and 40-31 since they are still open

B. Continue the procedure steps to check the remaining valves since 40-30 and 40-31 are closed

C. An operator must be sent to verify no flow from the vent overflow funnels servicing 40-30 and 40-31

D. Ensure the Core Spray inside isolation valves are closed since 40-30 and 40-31 position is unknown

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 41 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 41
System ID:	21365
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Question Derived from bank question.

O1-OPS-001-223-1-01, EO-1.5b, EO-1.7g

Answer: B: is correct - There is a note in SOP-40.2 that states that the absence of light indication for 40-30 and 40-31 is confirmation that these valves are closed. Since these valves are in the DW it is not possible to locally verify position. The breakers for these valves are locked open.

Distractor: A: is not correct - Since these valves are in the DW it is not possible to locally close them.

Distractor: C: is not correct - The valves are closed. No further action is required to ensure they are in the closed position.

Distractor: D: is not correct - Keeping inside core spray IVs closed is not necessary since the 40-30 and 40-31 valves are assumed to be closed.

REFERENCE: N1-SOP-40.2

REFERENCES PROVIDED TO THE CANDIDATE: NONE

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 41 Cross References (table item links)

10CFR55

- 41(b)(9)

Cognitive Level

- 1

NUREG 1123 KA Catalog Rev. 2

- 223002 A3.01 3.4/3.4 System indicating lights and alarms

Level of Difficulty

- Level 2: System operation and response

LP

- O1-OPS-001-223-1-01 Rev. na

Question Source

- Bank

PROC

- N1-SOP-40.2, Rev. NA

Question Setting

- C1 (License class closed reference)

42 **SYSID: 21366** **Points: 1.00**

With the plant operating at 100% power, a full main steam isolation valve closure occurs followed two seconds later by an automatic reactor scram on high neutron flux.

Assuming no operator action is taken, which one of the following is the operational implication of this event?

A. Relief valves open to maintain reactor pressure below the safety limit of 1250 psig. No safety valves open

B. Relief valves open to maintain reactor pressure below the safety limit of 1375 psig. No safety valves open

C. Relief valves and then safety valves open to maintain reactor pressure below the safety limit of 1250 psig

D. Relief valves and then safety valves open to maintain reactor pressure below the safety limit of 1375 psig

Answer: D

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 42 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 42
System ID:	21366
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-004-321-1-01, EO-3.0b

Answer: D: is correct - Per Technical Specification 2.2.1, the pressure safety limit of 1375 psig was derived from the design pressures and applicable codes for the reactor pressure vessel and the reactor coolant system piping. The ASME code permits pressure transients up to 10% over the design pressure (110% x 1250 psig design pressure = 1375 psig safety limit). The safety valves are sized according to the code for a condition of main steam isolation valve closure at 1850 mwth (100% power), followed by a reactor scram on high neutron flux. Under these conditions, a total of 9 safety valves are required to limit reactor pressure below the safety limit of 1375 psig.
Per SAR Chapter 15, for a full MSIV closure from 100% power (1850 mwth) and the reactor scrams on the MSIV closure signal, the reactor pressure would be maintained below the safety limit with ERVs, and no safety valves would be required to open. In this question, the reactor scrams on high neutron flux after the MSIV closure scram failed, requiring safety valve operation to limit the reactor pressure transient.

Distractor: A: is not correct - The safety limit is 1375 psig, not 1250 psig. 1250 psig is the design limit, not the safety limit which is 110 of the design limit. Safety valves must open to maintain pressure below the safety limit. Pressure would rise above 1250 psig but not 1375 psig.

Distractor: B: is not correct - Safety valves must open to maintain pressure below the safety limit.

Distractor: C: is not correct - The safety limit is 1375 psig, not 1250 psig. 1250 psig is the design limit, not the safety limit which is 110 of the design limit.

REFERENCE: TS 2.2.1 bases. UFSAR accident analysis.

REFERENCES PROVIDED TO THE CANDIDATE:
NONE

Question 42 Cross References (table item links)

10CFR55

- 41(b)(3)
- 41(b)(5)

Cognitive Level

- 1

FSAR

- UFSAR Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 239002 K5.02 3.7/3.8 Safety function of SRV operation

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-004-321-1-01, Rev. NA

Question Source

- New

TECHSPEC

- TS 2.2.1 bases, Rev. NA

Question Setting

- C1 (License class closed reference)

43 **SYSID: 21367** **Points: 1.00**

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

- 11 and 13 FW pumps are in service
- FWLC is in three-element control selected to Level Column 11
- Level Column 11 input to FWLC fails to zero over a thirty (second) period

Which one of the following describes the expected RPV level response and whether or not operator action is required to prevent an automatic reactor scram?

A. RISE and the operator must manually control 13 FCV to recover RPV level and prevent a reactor scram

B. LOWER and the operator must manually control 13 FCV to recover RPV level and prevent a reactor scram

C. RISE but system automatic response will recover RPV level. No operator action is required to prevent a reactor scram

D. LOWER but system automatic response will recover RPV level. No operator action is required to prevent a reactor scram

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 43 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 43
System ID:	21367
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-259-1-02, EO-1.4b, EO-1.4c, EO-1.8b

Answer: A: is correct - Level input low will cause a level error that will act to open 13 FCV and raise RPV level. Taking 13 FCV to manual will correct the problem. If manual action is not taken by the operator, the reactor will scram when the turbine trips on high reactor water level.

Distractor: B: is not correct - Level will rise not lower. If the level instrument failed high, then RPV water level would lower.

Distractor: C: is not correct - The system will not correct the malfunction automatically by swapping the selected level column. This is a manual operator action taken after level control is established in manual.

Distractor: D: is not correct - The system will not correct the malfunction automatically by swapping the selected level column. This is a manual operator action taken after level control is established in manual. Level will rise not lower. If the level instrument failed high, then RPV water level would lower.

REFERENCE: Ops Tech Chapter 23b, page 20

REFERENCES PROVIDED TO THE CANDIDATE:

NONE

Question 43 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 2

NUREG 1123 KA Catalog Rev. 2

- 259002 K6.05 3.5/3.5 Reactor water level input

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-259-1-02 Rev. na

OTHER_REFS

- OPS TECH, Rev. NA

Question Source

- New

Question Setting

- C1 (License class closed reference)

44 **SYSID: 21368** **Points: 1.00**

With the plant at 100% the following occur:

- Reactor Building Ventilation Exhaust Radiation Monitor 11 fails upscale and only RBEVS Train 11 automatically starts.
- RBEVS Train 12 fan cannot be started.

Assuming NO additional operator action is taken for RBEVS, which one of the following is an expected Reactor Building Differential Pressure and RBEVS flow twenty (20) minutes later as indicated on the RX BLDG/ATM DP in the control room?

A. 800 cfm and -0.25 inches WG

B. 800 cfm and +0.25 inches WG

C. 1600 cfm and -0.25 inches WG

D. 1600 cfm and +0.25 inches WG

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 44 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 44
System ID:	21368
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-261-1-01, EO-1.4c, EO-1.11a

Answer: C: is correct - Each train is designed to maintain the desired reactor building negative pressure. The failure of one train to start has no consequence to maintaining the design d/p of -0.25 inches WG. Each train will control flow at 1600 cfm regardless of the number of trains in service.

Distractor: A: is not correct - D/P will be maintained negative and flow through 11 maintained at 1600 cfm.

Distractor: B: is not correct - Flow through 11 RBEVS will be maintained at 1600 cfm.

Distractor: D: is not correct - Despite the loss of the 12 train, the remaining train will maintain D/P negative at 1600 cfm.

REFERENCE:
N1-OP-10, N1-ST-C5 (test one train while other train is in P-T-L)

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 44 Cross References (table item links)

10CFR55

- 41(b)(9)

Cognitive Level

- 1

NUREG 1123 KA Catalog Rev. 2

- 261000 A1.04 3/3.3 Secondary containment differential pressure

Level of Difficulty

- Level 2: System operation and response

LP

- O1-OPS-001-261-1-01 Rev. na

Question Source

- New

PROC

- N1-OP-10 Rev. NA
- N1-ST-C5 Rev. NA

Question Setting

- C1 (License class closed reference)

45 **SYSID: 21369** **Points: 1.00**

The plant is at 80% power. A grid disturbance causes the following conditions:

- A6-3-3, 345 KV SYS FREQUENCY HIGH-LOW
- A6-2-6, 345 KV BUS VOLTAGE HIGH-LOW
- A8-1-3, 115 KV BUS LOW VOLTAGE
- FREQUENCY at 58.8 hz, VOLTAGE at 113.6 v
- Computer points F432, F433, F434 indicate 113.6 volts
- Load Flow Computer NOT available
- The CRS has directed that EDG 103 be started and loaded

Which one of the following is the correct regarding the EDG 103 start based on the above conditions?

A. EDG103 is started locally and not from the control room

B. EDG103 is manually hand jacked before starting it for operation

C. PB103 is de-energized before closing the EDG103 output breaker

D. Alternate 125VDC feeds for EDG103 are aligned before starting it

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 45 Details

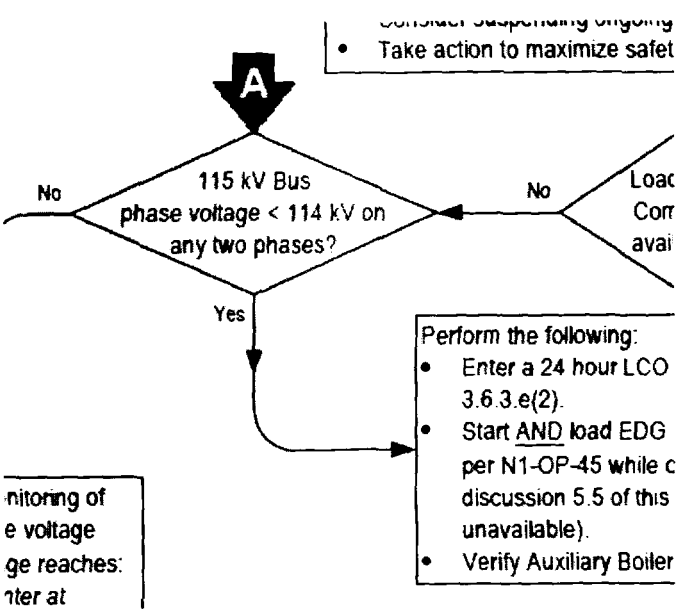
Question Type:	Multiple Choice
Topic:	RETAKE RO 45
System ID:	21369
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-264-1-01, EO-1.4c, EO-1.7e, EO-1.7g

Answer: C: is correct - This is a dead bus transfer because of the degraded grid conditions.

Distractor: A: is not correct - Started from the control room; but a dead bus transfer.

Distractor: B: is not correct - For the monthly surveillance, not for start for degraded grid conditions. This is a dead bus transfer.

Distractor: D: is not correct - DC power is not affected so normal feed can be used.



Question matches the K/A because with grid voltage at 113.6 volts the offsite circuits are considered inoperable per Tech Specs which is why EDG 103 must be started. The candidate must determine the correct method to start EDG 103 under these degraded conditions through an understanding of the interrelations of EDG 103 and the offsite circuits when in this degraded state. The method of starting EDG 103 is different under these conditions as compared to the surveillance run where the EDG is paralleled with the offsite circuit.

REFERENCE: N1-OP-45, Sect. E.3.0, N1-SOP-33A.3

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 45 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 2

NUREG 1123 KA Catalog Rev. 2

- 262001 A2.11 3.2/3.6; Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Degraded system voltages

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-264-1-01 Rev. NA

Question Source

- New

PROC

- N1-OP-45 Rev. NA
- N1-SOP-33A.3, Rev. NA

Question Setting

- C1 (License class closed reference)

46 **SYSID: 21370** **Points: 1.00**

The plant is on-line at 100% power with all electrical line-ups in their normal configuration, when the following transient occurs:

- 115 KV line voltage is lost
- The undervoltage situation is sustained to greater than 30 seconds
- Breakers R10 & R40 open
- All protective relaying at Lighthouse Hill operates as designed
- The available generator at Bennetts Bridge Auto transfers and energizes the line from Bennetts Bridge to Lighthouse Hill

Which one of the following is the anticipated response?

A. Line # 1 is energized to J.A. Fitzpatrick and to NMP1 and breaker R10 recloses to energize the 115KV bus

B. Line # 1 is energized to J.A. Fitzpatrick and to NMP1 and breaker R40 recloses to energize the 115KV bus

C. Line # 4 is energized to J.A. Fitzpatrick and to NMP1 and breaker R40 recloses to energize the 115KV bus

D. Line # 4 is energized to J.A. Fitzpatrick and to NMP1 and breaker R10 recloses to energizes the 115KV bus

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 46 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 46
System ID:	21370
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Question Derived from bank question.

O1-OPS-001-262-1-01, EO-1.4,EO-1.5, EO-1.7, EO-1.8

Answer: C: is correct - Because LightHouse Hill feeds line 4 thru the JAF switchyard and auto reclosure, sensing the voltage on line 4 recloses R40.

Distractor: A: is not correct - Because line 1 is energized from Oswego not LightHouse Hill.

Distractor: B: is not correct - Because line 1 is energized from Oswego not LightHouse Hill, and R40 is on line 4.

Distractor: D: is not correct - Because breaker R10 is on line 1 from Oswego.

REFERENCE: N1-OP-33A, Pages 3 and 4

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 46 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 1

NUREG 1123 KA Catalog Rev. 2

- 262001 K1.03 3.4/3.8 Off-site power sources

Level of Difficulty

- Level 2: System operation and response

LP

- O1-OPS-001-262-1-01 Rev. NA

Question Source

- Bank

PROC

- N1-OP-33A Rev. NA

Question Setting

- C1 (License class closed reference)

47 **SYSID: 21371** **Points: 1.00**

The plant is operating at 100% power. UPS 172A is in service when its inverter fails and its output goes to zero. Which one of the following is the effect on RPS 12 Logic power?

A. Energized from UPS 172B without a loss of power or manual operator action

B. Energized from UPS 172B following a loss of power and manual operator action

C. Energized from 125 VDC BB 12 without a loss of power or manual operator action

D. Energized directly from PB 17 Section B without a loss of power or manual operator action

Answer: D

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 47 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 47
System ID:	21371
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-212-1-01, EO-1.8b

Answer: D: is correct - Power is automatically transferred to the Bypass Step Down Transformer on a loss of the inverter.

Distractor: A: is not correct - This requires a manual transfer and loss of bus RPS Logic power.

Distractor: B: is not correct - Logic power will NOT be lost because the Bypass will pick up the logic.

Distractor: C: is not correct - The DC power cannot power the logic because the inverter has failed.

REFERENCE: N1-OP-40, Section B

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 47 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 1

NUREG 1123 KA Catalog Rev. 2

- 262002 A3.01 2.8/3.1; Ability to monitor automatic operations of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) including: Transfer from preferred to alternate source

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-212-1-01 Rev. na

Question Source

- New

PROC

- N1-OP-40 Rev. NA

Question Setting

- C1 (License class closed reference)

48 **SYSID: 21372** **Points: 1.00**

The plant is at 100% power when A3-4-3, BATTERY BOARD 12 GROUND, alarms. The ground is confirmed using the Panel A voltmeter.

Per N1-OP-47A, H.8.0, Locating a Ground in the 125vdc Power System, to expedite locating the ground which one of the following ground isolation actions is performed FIRST?

A. Check logs to determine if DC equipment was recently started and if not then open SBC 171 A-B DC breaker and check to see if the ground clears

B. Check logs to determine if DC equipment was recently started and if not then remove Battery 12 from service and check to see if the ground clears

C. Remove Battery 12 from service and check to see if the ground clears and if it is still present then check logs to determine if DC equipment was recently started

D. Open SBC 171 A-B DC breaker and check to see if the ground clears and if it is still present then remove Battery 12 from service and check to see if the ground clears

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 48 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 48
System ID:	21372
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-263-1-01, EO-1.7d

Answer: A. is correct - The operator actions performed are as follow: First action is to determine if anything done recently (i.e., start plant equipment) that could have caused the ground. If not then the SBC is disconnected. If not the SBC then the Battery is disconnected.

Distractor: B: is not correct - SBC is checked before disconnecting the Battery.

Distractor: C: is not correct - Check the logs first, and then initiate isolation actions.

Distractor: D: is not correct - Check the logs first, and then initiate isolation actions.

Question matches the K/A because for two part K/As, if both parts are not evaluated then the more important of the two parts is to be evaluated. For this K/A, performing the ground isolation and to locate and isolate the affected component is more important than just predicting that grounds can lead to a loss of the affected DC system (i.e., loss of Battery Board 11, or a fire, or both). The question evaluates the importance and precedent for actions outlined in the operating procedure. If the steps are not performed in the order specified, unnecessary entry into TS actions and unnecessary de-energization of plant equipment could occur. Per N1-OP-47A. D.8.0: It is important to expedite trouble shooting and repair of a ground on the DC system because multiple grounds could cause

failure or initiation of safety equipment.

REFERENCE: N1-OP-47A, H.8.0

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 48 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 1

NUREG 1123 KA Catalog Rev. 2
- 263000 A2.01 2.8/3.2 Grounds, Rev. NA

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-001-263-1-01 Rev. NA

Question Source
- New

PROC
- N1-OP-47A Rev. NA

Question Setting
- C1 (License class closed reference)

49 **SYSID: 21373** **Points: 1.00**

Given the following conditions during the performance of N1-ST-M4A, Emergency Diesel Generator 102 Operability Test:

- EDG 102 was just paralleled with offsite power and loaded to 500 KW.
- EDG 102 KVAR indicates zero (0).

Which one of the following EDG 102 switch adjustments is required to establish the correct KVAR loading under the above conditions?

A. Place the DIESEL GOV control switch to RAISE to pick up leading vars

B. Place the DIESEL GOV control switch to LOWER to pick up lagging vars

C. Place the VOLT ADJ RHEO GEN 102 switch to RAISE to pick up lagging vars

D. Place the VOLT ADJ RHEO GEN 102 switch to LOWER to pick up leading vars

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 49 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 49
System ID:	21373
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-264-1-01, EO-1.4b, EO-1.7f

Answer: C: is correct - Per N1-ST-M4A, 8.1.16.f: If performing N1-ST-M4A alone adjust VOLT ADJ RHEO GEN 102 switch to establish reactive load between 300 and 800 KVARS. The switch must be place to RAISE to pick up lagging vars.

Distractor: A: is not correct - This is the wrong control switch and it is used for speed control. Has little effect on reactive loading.

Distractor: B: is not correct - This is the wrong control switch and it is used for speed control. Has little effect on reactive loading.

Distractor: D: is not correct - Must place the switch to RAISE to pick up lagging vars (300-800 KVARs).

REFERENCE:
N1-ST-M4A, 8.1.16.f.

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 49 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 1

NUREG 1123 KA Catalog Rev. 2

- 264000 A4.01 3.3/3.4 Adjustment of exciter voltage

Level of Difficulty

- Level 2: System operation and response

LP

- O1-OPS-001-264-1-01 Rev. NA

Question Source

- New

PROC

- N1-ST-M4A, Rev. NA

Question Setting

- C1 (License class closed reference)

50 **SYSID: 21374** **Points: 1.00**

EDG 102 has received an automatic start signal but has failed to reach 200 rpm in 5 seconds.

Which one of the following describes the automatic response of the DG and the manual operator actions required to start EDG 102?

A. EDG 102 shuts down immediately. After correcting the start failure condition, place the engine control switch to STOP and then manually restart EDG 102

B. EDG 102 attempts a second start. If this start fails, after correcting the start failure condition, place the engine control switch to STOP and then restart manually EDG 102

C. EDG 102 shuts down immediately. After correcting the start failure condition, depress the local 48X and ALARM RESET AND FAST STOP pushbuttons and then verify restart EDG 102

D. EDG 102 attempts a second start. If this start fails, after correcting the start failure condition, depress the local ALARM RESET AND FAST STOP and the local 48X pushbuttons and then verify restart EDG 102

Answer: D

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 50 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 50
System ID:	21374
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Question Derived from bank question.

O1-OPS-001-264-1-01, EO-1.4b, EO-1.7e, EO-1.8b

Answer: D. is correct - If the EDG fails to reach 200 RPM in 5 seconds, a second start is attempted. After the EDG fails to reach 200 RPM in 5 seconds on the second start attempt – this startup attempt continues for another 2 minutes before it is terminated. Before another attempt can be made, the fault condition must be corrected and the 48X locally reset.

Distractor: A: is not correct - Second start is automatically attempted before shutdown. Not the correct action to establish conditions to restart the EDG.

Distractor: B: is not correct - Not the correct action to establish conditions to restart the EDG.

Distractor: C: is not correct - Second start is automatically attempted before shutdown.

REFERENCE: Ops Tech Chapter 34 pages 35-36, N1-ARP-A4 A4-2-3

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 50 Cross References (table item links)

10CFR55

- 41(b)(7)

Cognitive Level

- 2

NUREG 1123 KA Catalog Rev. 2

- 264000 K1.06 3.2/3.2 Starting system

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-264-1-01 Rev. NA

OTHER REFS

- OPS TECH, Rev. NA

Question Source

- Bank

PROC

- N1-ARP-A4, Rev. NA

Question Setting

- C1 (License class closed reference)

51 **SYSID: 21375** **Points: 1.00**

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

- IAC #11 is the LEAD compressor
- IAC #12 is the backup compressor
- IAC #13 is clearance tagged for maintenance
- Inter-Tie BV 94-91 is in MANUAL and is open
- Subsequently, PB 16A is de-energized due to an electrical fault

Without any operator action, which one of the following is the resultant Instrument Air System (IAS) pressure and availability to Non-Safety Related (NSR) loads?

	IAS PRESSURE	AVAILABLE TO NSR LOADS
A.	101-106 psig	Yes
B.	101-106 psig	No
C.	96-102 psig	Yes
D.	96-102 psig	No

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 51 Details

Question Type:	*Multiple Choice*
Topic:	RETAKE RO 51
System ID:	21375
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Question Derived from bank question.

O1-OPS-001-278-1-01, EO-1.4b, EO-1.4c, EO-1.5b

Answer:	C: is correct - Loss of PD 16A results in loss of #11 IAC, which is selected as the lead compressor. #12 IAC with the selection switches in position "2" will start and load when pressure drops to 96 psig, then cycle between 96 – 102 psig. With BV-94-91 open and in manual, air will be maintained to the NSR loads.
Distractor:	A: is not correct - These are the lead compressor setpoints (#11).
Distractor:	B: is not correct - These are the lead compressor setpoints and BV-94-91 remains open.
Distractor:	D: is not correct - BV-94-91 remains open.

REFERENCE: N1-OP-20, Section B

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 51 Cross References (table item links)

10CFR55

- 41(b)(7)
- 41(b)(9)

Cognitive Level

- 2

NUREG 1123 KA Catalog Rev. 2

- 300000 K2.01 2.8/2.8 Instrument air compressor

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-278-1-01 Rev. na

Question Source

- Bank

PROC

- N1-OP-20 Rev. NA

Question Setting

- C1 (License class closed reference)

52 **SYSID: 21376** **Points: 1.00**

The plant is operating, with the following:

- TBCLC is in a normal lineup
- 11 TBCLC pump is running
- 12 and 13 TBCLC heat exchangers are in service
- Condenser intake temperature is 38°F
- A loss of instrument air to TBCLC temperature controller TC-71-20 occurs

Which one of the following is the expected change in the TBCLC supply temperature and the action to be taken to control this temperature?

	Temp Response	Action To Control
A.	Lower	Manually control TCV 71-88 TBCLC TCV and TCV, 72-147 SW TCV
B.	Lower	Throttle SW Drag valve bypass 72-93R BYPASS AROUND TURB BLDG SW DRAG VALVE
C.	Rise	Manually control TCV 71-88 TBCLC TCV and TCV, 72-147 SW TCV
D.	Rise	Throttle SW Drag valve bypass 72-93R 93R BYPASS AROUND TURB BLDG SW DRAG VALVE

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 52 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 52
System ID:	21376
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Question Derived from bank question.

O1-OPS-001-274-1-01, EO-1.5b

Answer: A: is correct - On a loss of control air signal from E/P 71-20, HX TCV 71-88 will fail as is and SW TCV 72-147 will fully open fully maximizing cooling. With SW flow maximized at this low a lake temperature, heat exchanger outlet temperature will lower. To raise the heat exchanger outlet temperature, take manual control of SW TCV and/or HX TCV 71-88 and operate the valves to raise temperature.

With cooling maximized supply temperature will lower. Distracters are combinations of incorrect response or incorrect action. 72-93R is normally closed, opening 72-93R will lower temperature).

Distractor: B: is not correct - 72-93R is normally closed, opening 72-93R will lower temperature).

Distractor: C: is not correct - With cooling maximized supply temperature will lower.

Distractor: D: is not correct - With cooling maximized supply temperature will lower. 72-93R is normally closed, opening 72-93R will lower temperature).

Question matches the K/A because the candidate must determine the effect of the loss of control signal and the resultant temperature change (high/low CCW temp), and then determine the correct valve manipulation (action) to correct/return to TBCLC temperature to its pre-transient value. Both parts of the two part K/A are evaluated.

REFERENCE: N1-OP-24, H.9.0, H.11.0

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 52 Cross References (table item links)

10CFR55
- 41(b)(4)

Cognitive Level
- 2

NUREG 1123 KA Catalog Rev. 2
- 400000 A2.03 2.9/3 High/low CCW temperature

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-001-274-1-01 Rev. na

Question Source
- Bank

PROC
- N1-OP-24 Rev. NA

Question Setting
- C1 (License class closed reference)

53 **SYSID: 21377** **Points: 1.00**

The plant is operating at 100% power when the following occur:

- H1-4-1, R BUILDING COOLING WATER PRESS TEMP MAKEUP FLOW, in alarm
- B108, RB COOL WTR PMPS HDR PR LOW, in alarm
- The standby RBCLC pump is started; RBCLC pressure is constant at 39 psig
- The plant operator reports no flow from the RBCLC high point vent

Which one of the following is the correct action in response to the above conditions?

A. Immediately enter N1-SOP-11.1, RBCLC Failure

B. Perform N1-OP-11, H.16.0, Internal System Leakage

C. Place a third RBCLC heat exchanger in service per N1-OP-11

D. Immediately scram the reactor and enter N1-SOP-1, Reactor Scram

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 53 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 53
System ID:	21377
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-208-01, EO-1.7d, EO-1.7g

Answer: A: is correct - Per ARP H1-4-1: If header pressure low, the perform the following:
a. Start Standby pump per N1-OP-11, Section F.
b. If system pressure drops to 40 psig, THEN enter N1-SOP-11, RBCLC Failure and execute concurrently.

Distractor: B: is not correct - This would be required if there was flow from the RBCLC high point vent. With no flow from the vent, internal system leakage is not a concern.

Distractor: C: is not correct - If pressure can be restored to normal (~60 psig) either shut down standby RBCLC pump, or place third Heat Exchanger in service per N1-OP-11. Not required at this time since personnel efforts are to be applied to restoring header pressure.

Distractor: D: is not correct - The first action would be to perform an emergency power reduction to provide time to determine the cause of the low pressure. If the cause could not be determined and corrected, or if RBCLC temperature begins to rise, then a reactor scram could be directed by the SM/CRS.

REFERENCE: N1-ARP-H1 H1-4-1, N1-SOP-11.1

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 53 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 1

NUREG 1123 KA Catalog Rev. 2
- 400000 A1.03 2.7/2.7 CCW Pressure

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-001-208-1-01 Rev. na

Question Source
- New

PROC
- N1-ARP-H1 Rev. NA
- N1-SOP-11.1, Rev. NA

Question Setting
- C1 (License class closed reference)

54 **SYSID: 21378** **Points: 1.00**

The plant operating at power, with the following:

- A control rod is initially at position 12
- This control rod's HCU Insert Supply Valve (SOV 123) is stuck closed
- NO other Control Rod Drive Hydraulic system failures occur

Which one of the following describes the resulting control rod movement in response to separate single notch INSERT and single notch WITHDRAW commands (from position 12)?

	Single Notch INSERT Command Response	**Single Notch WITHDRAW Command Response**
A.	Inserts to 00	Withdraws to 14
B.	Inserts to 10	Remains at 12
C.	Remains at 12	Remains at 12
D.	Remains at 12	Withdraws to 14

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 54 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 54
System ID:	21378
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-201-1-02, EO-1.4b, EO-1.8b

Answer: C: is correct - Per N1-OP-5, H.13.0 Notes:
If drive water flow does not increase, or increases then returns to zero, it could be an indication of a Directional Control Valve failure.
When attempting to INSERT the control rod one notch, drive water flow does NOT change and remains at 0.0 gpm. When attempting to withdraw the same control rod, drive water flow increases to 2.0 gpm then returns to 0.0 gpm. The control rod does NOT move inward. The control rod has to momentarily insert to cam out the collet fingers when the withdrawal signal is applied but the insert signal cannot be applied.

Distractor: A: is not correct - Control rod does not move to next notch.

Distractor: B: is not correct - Control rod does not move to next notch.

Distractor: D: is not correct - Control rod does not move to next notch.

REFERENCE:
N1-OP-5, H.13.0 notes and Ops Tech Chapter 5a page 17 and 18

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 54 Cross References (table item links)

10CFR55

- 41(b)(6)

Cognitive Level

- 2

NUREG 1123 KA Catalog Rev. 2

- 201001 K5.08 2.5/2.6 Solenoid operated valves

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-201-1-02 Rev. na

OTHER_REFS

- OPS TECH, Rev. NA

Question Source

- New

PROC

- N1-OP-5 Rev. NA

Question Setting

- C1 (License class closed reference)

55 **SYSID: 21379** **Points: 1.00**

The plant is at 85% power with the following conditions observed for Reactor Recirc Pump (RRP) 15:

- RRP 15 HP Seal Pressure is 1030 psig
- RRP 15 LP Seal Pressure is 1000 psig
- A142 RRP 15 SEAL LEAK FLOW HIGH in alarm
- A144 RRP 15 SEAL LEAK DETECTION FLOW HIGH in alarm
- F2-1-5, REACTOR RECIRC PUMP-MOTOR 15 in alarm
- Drywell pressure has risen from 1.2 to 1.3 psig as a result of the above conditions

Based on the above conditions, which one of the following is the correct evaluation of the RRP 15 seal degradation including a prediction of the magnitude of the failure (catastrophic or not catastrophic)?

A. Degradation of the HP seal. NO degradation of the LP seal. Failure is classified as catastrophic at this time.

B. Degradation of the LP seal. NO degradation of the HP seal. Failure is NOT classified as catastrophic at this time.

C. Degradation of both the LP and HP Seals. Failure is classified as catastrophic at this time

D. Degradation of both the LP and HP Seals. Failure is NOT classified as catastrophic at this time.

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 55 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 55
System ID:	21379
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-202-1-01, EO-1.7e, EO-1.7g

Answer: C: is correct - Normal HP seal cavity pressure is approximately equal to reactor pressure. Normal LP seal cavity pressure is approximately 500 psig. An HP seal failure results in the LP seal cavity pressure approaching the HP seal cavity pressure of 1000 psig. To distinguish between just degradation of one seal or both, the candidate must consider the computer points in alarm. With both seal cavity pressures at full reactor pressure, and with both the seal leak flow high and seal leak detection flow high in alarm, degradation has occurred on both seals. If the computer points indicated were not in alarm, then it would be concluded that only one of the two seals (HP seal) was degraded, and not both.
Per N1-SOP-1.2, 5.3: A catastrophic failure of a Reactor Recirculation Pump seal is defined as any Recirc Pump seal failure that results in a noticeable increase in Drywell pressure or Drywell floor drain leakage. This is indicative a small break LOCA.

Distractor: A: is not correct - LP seal is also degraded.

Distractor: B: is not correct - HP seal is also degraded. Because of the noticeable drywell pressure increase, the failure is

considered to be catastrophic.

Distractor: D: is not correct - Because of the noticeable drywell pressure increase, the failure is considered to be catastrophic.

Question matches the K/A because the candidate is required to evaluate monitored changes in the reactor recirc pump seal pressures and determine the failed seals, and also classifying the magnitude of the seal failure which is a prediction based on the conditions presented. As a result of this determination, operation of the correct reactor recirc controls associated with this failure would be taken as outlined in N1-SOP-1.2.

REFERENCE:
N1-SOP-1.2, 5.3.
Ops Tech Chapter 4

REFERENCES PROVIDED TO THE CANDIDATE: NONE

MODIFIED: The original bank question just provided indications and alarms and asked what seal(s) are failed for a failure of both seals. The question was modified by adding the effect of the conditions on drywell pressure (which increased) and requires the candidate to not only determine the failed seal(s) but also classify the magnitude of the seal failures. Based on the rising drywell pressure the candidate is required to determine that the magnitude of the seal failures is considered catastrophic. If drywell pressure was unchanged, then the magnitude of the seal failure is not considered catastrophic.

Question 55 Cross References (table item links)

10CFR55
- 41(b)(3)

Cognitive Level
- 2

NUREG 1123 KA Catalog Rev. 2
- 202001 A1.09 3.3/3.3 Recirculation pump seal pressures

Level of Difficulty
- Level 3: Higher order Knowledge item

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

OTHER_REFS
- OPS TECH, Rev. NA

Question Source
- Modified

PROC
- N1-SOP-1.2 Rev. NA

Question Setting
- C1 (License class closed reference)

56 **SYSID: 21380** **Points: 1.00**

The plant is at 100% power when an event occurs, with the following conditions:

- Reactor Water Cleanup (RWCU) System is in service
- Power Board 16B is de-energized
- All running RBCLC pumps trip

Which one of the following describes the impact on RWCU?

A. RWCU pumps trip and the system isolates, BV-33-02R, CU SUPPLY ISOLATION VALVE 11 (INSIDE) fails to close

B. Remains running, Manual RWCU isolation required, BV-33-01R, CU RETURN ISOLATION VALVE 1 (INSIDE) fails to close

C. RWCU pumps trip and the system isolates, BV-33-01R, CU RETURN ISOLATION VALVE 1 (INSIDE) fails to close

D. Remains running, manual RWCS isolation required, BV-33-02R, CU SUPPLY ISOLATION VALVE 11 (INSIDE) fails to close

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 56 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 56
System ID:	21380
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-204-1-01, EO-1.5b

Answer: A: is correct - Loss of the RBCLC pumps and flow will cause NRHX outlet temperature to exceed 140°F immediately, which results in an automatic isolation signal for RWCU. The pumps are interlocked with the IV such that the pumps will trip when the system isolates. Loss of power on PB16B(161B) prevents closure of BV 33-02R.

Distractor: B: is not correct - Auto isolation signal is received as explained above. BV-33-01R is powered from PB17B(171B) and will close when the isolation signal is received.

Distractor: C: is not correct - BV-33-01R will close upon receipt of the auto isolation signal because it is powered from PB17B(171B).

Distractor: D: is not correct - An automatic system isolation is received as explained in "A" above.

REFERENCE: N1-OP-3 RWCU Pages 8 and 207

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 56 Cross References (table item links)

10CFR55

- 41(b)(5)

Cognitive Level

- 2

DRW

- C-19437-C Rev. NA
- C-19859-C Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 204000 A3.03 3.6/3.6 Response to system isolations

Level of Difficulty

- Level 2: System operation and response

LP

- O1-OPS-001-204-1-01 Rev. na

Question Source

- New

PROC

- N1-OP-3 Rev. NA
- N1-SOP-11.1, Rev. NA

Question Setting

- C1 (License class closed reference)

57 **SYSID: 21381** **Points: 1.00**

Given the following while executing the EOPs:

- Operation is within the BAD REGION of Figure B, RPV SATURATION TEMPERATURE
- The Fuel Zone digital displays are NOT flashing

Which one of the following is the relationship between INDICATED and ACTUAL RPV water level for on-scale level instruments?

A. All instruments except Fuel Zone indicate LOWER than actual level.

B. All instruments except Fuel Zone indicate HIGHER than actual level.

C. All instruments including Fuel Zone indicate LOWER than actual level.

D. All instruments including Fuel Zone indicate HIGHER than actual level.

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 57 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 57
System ID:	21381
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Question Derived from bank question.

O1-OPS-006-344-1-02, EO-1.2, EO-1.3, EO-1.5b

Answer: b. is correct - The RPV SATURATION TEMPERATURE is a plot of the saturation temperature of water as a function of pressure. If the temperature of the water in an RPV water level instrument run exceeds this temperature, the water may start to boil, resulting in unreliable indication. Boiling is of concern in all instrument runs – horizontal and vertical, reference and variable, drywell and containment. Boil off from the reference leg reduces the height of water in the leg. This decreases the pressure on the reference leg side of the DP cell and increases the indicated level. Boiling in the variable leg increases the pressure on the variable leg side of the DP cell, likewise increasing the indicated level. Continued boiling would produce an increasing level trend.
The Fuel Zone instruments are density compensated, and will provide an accurate indication of level when in the BAD REGION as long as it the digital Fuel Zone indicators are not flashing indicating instrument run boiling.

Distractor: A: is not correct - Affected

instruments will indicate higher than actual level, not lower than actual level.

Distractor: C: is not correct - Fuel zone instruments are not directly affected by operation within the BAD REGION of Figure B. Affected instruments will indicate higher than actual level, not lower than actual level.

Distractor: D: is not correct - Fuel zone instruments are not directly affected by operation within the BAD REGION of Figure B.

REFERENCE:
NER-1M-095, 1.2.C.

REFERENCES PROVIDED TO THE CANDIDATE:
N1-EOP-2

Question 57 Cross References (table item links)

10CFR55
- 41(b)(7)

Cognitive Level
- 1

NUREG 1123 KA Catalog Rev. 2
- 216000 K5.07 3.6/3.8 Elevated containment temperature effects on level indication

Level of Difficulty
- Level 2: System operation and response

LP
- O1-OPS-006-344-1-02, Rev. NA

OTHER_REFS
- (REPORT) NER-1M-095, Rev. NA

Question Source
- Bank

PROC
- N1-EOP-2 Rev. NA

Question Setting
- C1 (License class closed reference)

58 **SYSID: 21382** **Points: 1.00**

The plant is operating at power, with the following:

- A Stuck Open ERV event is in progress
- Containment Spray Loop 121 is being placed in Torus Cooling using the EOP
- At the step to open valve 80-118, CONT SPRAY TEST TO TORUS FCV, valve 80-40, CONT SPRAY BYPASS BV 111 is mistakenly opened
- THEN, Containment Spray pump #121 is started

Which one of the following conditions results from these actions?

A. Torus cooling through Loop 111 and 121.

B. Torus cooling through Loop 111, not Loop 121.

C. Containment spray through Loop 111, not Loop 121.

D. Containment spray pump 121 trips and Loop 111 must be used.

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 58 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 58
System ID:	21382
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Question Derived from bank question.

O1-OPS-006-344-1-04, EO-1.3, EO-1.4

Answer: C: is correct - Opening 80-40 rather than 80-118 directs Containment Spray Pump 121 flow into the drywell sprays through loop 111.

Distractor: A: is not correct - Drywell is sprayed. No torus cooling occurs.

Distractor: B: is not correct - Drywell is sprayed. No torus cooling occurs.

Distractor: D: is not correct - Drywell is sprayed. The containment spray pump does not trip.

REFERENCE:
EOP-1, Attachment 16, Torus Cooling, P&ID C-18012C, Sheet 2

REFERENCES PROVIDED TO THE CANDIDATE:
P&ID C-18012C, Sheets 1 and 2

Question 58 Cross References (table item links)

10CFR55

- 41(b)(8)

Cognitive Level

- 2

DRW

- C-18012-C Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 219000 A1.08 3.7/3.6 System lineup

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-006-344-1-04 Rev. na

Question Source

- Bank

PROC

- N1-EOP-1 Rev. NA

Question Setting

- C1 (License class closed reference)

59 **SYSID: 21383** **Points: 1.00**

With the plant at 100% power, the containment spray system controls are aligned for standby operation per N1-OP-14. Subsequently, a large break LOCA occurs with a loss of PB102.

Which one of the following is the design response of containment spray to provide rated spray flow in response to the above conditions?

A. Less than rated spray flow because only one pump in each loop discharges into the primary and secondary loops providing 50% flow in each.

B. Less than rated spray flow because one pump is required to provide a water seal while the other discharges into the primary or secondary loops.

C. Rated spray flow is achieved because two 50% capacity pumps discharge into the primary or secondary loop providing 100% flow in that loop.

D. Rated spray flow is achieved because two pumps are operating with only one required to provide full sprays from the primary and secondary loop.

Answer: D

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 59 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 59
System ID:	21383
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-226-1-01, EO-1.2, EO-1.5b

Answer: D. is correct - Each containment spray pump is rated for 100% flow. An open cross-tie valve on loops 111 and 122 insure that either of these pumps operating will supply the other header (Pumps 111 and 112 – supplied by PB 102 feeds the primary loop, Pumps 121 and 122 supplied by PB 103 feeds the secondary loop). Therefore either of the pumps into two loops would provide full rated spray flow.
The Containment Spray System is normally in standby, lined up for automatic start. It is initiated by two, simultaneous signals: high Drywell pressure at 3.5 psig, AND a low-low Reactor level at 5 inches. At twenty-five seconds after the initiation signal, pumps 111 and 121 start, and at thirty seconds pumps 112 and 122 start. If the Core Spray and Core Spray Topping Pumps receive initiation signals but fail to start, and the initiation signals are still present, then a set of backup timer contacts will cause the Containment Spray to start sequentially after a delay of fifty five seconds for Pumps 111 and 121, and sixty seconds for Pumps 112 and 122. There are no valve operations required for automatic initiation of the Containment spray System; self-actuated check valves open when sufficient discharge pressure is

developed by the pumps to deliver flow to the Containment.The Containment Spray System consists of two independent, automatically-activated 100% capacity loops, each capable of supplying full spray flow to the containment. Each loop consists of two suction headers, two Containment Spray Pumps, two Heat Exchangers, two Raw Water Pumps, a test line, three spray spargers, and associated piping and control valves. Operation of two Containment Spray Pumps is required to maintain the Appendix J Containment Spray Water Seal. Flow from one Containment Spray Pump (3600 GPM) (95% to the Drywell and the remainder to the Torus) is sufficient to remove the postulated post-accident core energy released. This includes energy released subsequent to an extensive metal-water reaction in the event of Core Spray failure.

Distractor: A: is not correct - Each pump is rated at 100% of the required containment spray flow.

Distractor: B: is not correct - Each pump is rated at 100% of the required containment spray flow and would discharge into both loops.

Distractor: C: is not correct - Each pump is rated at 100% , and they would spray into both loops (open cross-ties)

Question matches the K/A because signaling and relaying for automatic initiation are part of the controls and actuate in response to the LOCA. Also the controls are aligned manually to the standby condition before the automatic initiation is received. The candidate must predict the system flow in response to the LOCA signal with a loss of an emergency bus also.

REFERENCE: N1-OP-14, Section B, P&ID C-18012-C. Sheet 2

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 59 Cross References (table item links)

10CFR55

- 41(b)(8)

Cognitive Level

- 1

DRW

- C-18012-C Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 226001 A1.06 3.2/3.2 System flow

Level of Difficulty

- Level 2: System operation and response

LP

- O1-OPS-001-226-1-01 Rev. na

Question Source

- New

PROC

- N1-OP-14 Rev. NA

Question Setting

- C1 (License class closed reference)

60 **SYSID: 21384** **Points: 1.00**

During a refueling outage, the Phase 1 fuel shuffle is in progress:

- The next step (Step 112) requires that a fuel assembly be removed from the reactor core and be transported to and stored in its assigned spent fuel pool location.
- With the Step 112 fuel bundle grappled and in the fuel transfer canal (cattle chute) both refueling bridge air hoses rupture.

Which one of the following describes the effect of this failure on completing Step 112 using the LEFT HAND and RIGHT HAND console controls in the operator cab?

A. The refueling bridge motion is interrupted and it cannot be moved from its current position.

B. The refueling bridge can be moved into the spent fuel pool but the trolley cannot be moved left and right.

C. The fuel assembly can be positioned over its assigned location but the main hoist cannot be lowered.

D. The fuel assembly can be seated into its assigned location but the grapple hooks cannot be opened.

Answer: D

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 60 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 60
System ID:	21384
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-245-1-02, EO-1.5b

Answer: D. is correct - Air supplies the motive force to operate the grapple hooks. On a loss of electrical power the grapple hooks fail closed. On a loss of air the grapple hooks fail as is. With grapple hooks already closed and no air, they cannot be opened once the fuel assembly is seated.

Distractor: A: is not correct - The refueling bridge and trolley motion are electrical, not pneumatic.

Distractor: B: is not correct - The refueling bridge and trolley motion are electrical, not pneumatic.

Distractor: C: is not correct - The main hoist raise and lower motion is electrical, not pneumatic.

REFERENCE:
Ops Tech Chapter 37B

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 60 Cross References (table item links)

10CFR55

- 41(b)(13)

Cognitive Level

- 2

NUREG 1123 KA Catalog Rev. 2

- 234000 K6.04 2.9/3.7 Refueling platform air system: Plant-Specific

Level of Difficulty

- Level 2: System operation and response

LP

- O1-OPS-001-234-1-02 Rev. na

OTHER REFS

- (REPORT) NER-1M-095, Rev. NA

Question Source

- New

PROC

- N1-ARP-A2 Rev. NA
- N1-OP-32 Rev. NA

Question Setting

- C1 (License class closed reference)

61 **SYSID: 21385** **Points: 1.00**

Given the following conditions:

- The plant is starting up following a refueling outage
- Reactor power is 38%
- F3-4-6, FIRST STAGE BOWL PRESSURE LOW, in alarm
- A2-3-2, GENERATOR CORE MONITOR, alarms
- Operator presses and holds the core monitor FILTER pushbutton on PNL 4B, and observes the core monitor indication returns to normal (90%)

In response to the above conditions, which one of the following identifies the validity of the core monitor (valid or invalid) and the correct action to be taken per ARP A2-3-2?

A. Alarm is INVALID. Correct the improper gas flow before continuing the startup.

B. Alarm is INVALID. Correct the instrument malfunction while continuing the startup.

C. Alarm is VALID. Trip the main turbine and enter SOP-31.1, Turbine Trip.

D. Alarm is VALID. Scram the reactor and enter N1-SOP-1, Reactor Scram.

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 61 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 61
System ID:	21385
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-245-1-02, EO-1.5a, EO-1.5b, EO-1.7e

Answer: C: is correct - Per N1-ARP A2, A2-3-2:
Press and hold the "FILTER" pushbutton on PNL 4B, and observe the indication:
IF indication returns to near normal (90%), Alarm is valid and is caused by overheating. It is permissible to reduce Generator load while monitoring other generator parameters (no benefit since already low in power) and depress RESET pushbutton at Panel 4B to determine if the alarm can be cleared by the power reduction (no benefit since already low in power). The main generator must be removed from service as follows:
IF Rx power greater than 40% OR annunciator F3-4-6 is not in alarm, Manually Scram Reactor and enter N1-SOP-1 Reactor Scram. IF Rx power is less than 40%, then trip the turbine and enter N1-SOP-31.1, Turbine Trip.
Per N1-OP-32, D.5.0: The generator should not be operated with the core monitor in an alarm condition.

Distractor: A: is not correct - Alarm is valid, not invalid. Low gas flow is a possible cause of an invalid alarm (low or no flow) and is determined by checking gas flow

locally, not at PNL 4B.

Distractor: B: is not correct - Alarm is valid, not invalid. If the alarm was invalid, and does not clear after pressing RESET at PNL 4B, then the possible cause of the invalid alarm is an instrument failure. Should hold power until corrected, but not required.

Distractor: D: is not correct - Based on the reactor power level, only a main turbine trip is required.A reactor scram is not required.

REFERENCE: N1-ARP A2, A2-3-2; N1-OP-32, D.5.0

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 61 Cross References (table item links)

10CFR55
- 41(b)(5)

Cognitive Level
- 2

NUREG 1123 KA Catalog Rev. 2
- 245000 K5.07 2.6/2.9 Generator operations and limitations

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-001-245-1-02 Rev. NA

Question Source
- New

PROC
- N1-ARP-A2 Rev. NA
- N1-OP-32 Rev. NA

Question Setting
- C1 (License class closed reference)

62 **SYSID: 21386** **Points: 1.00**

With the plant operating at 100% power, the following occur:

- H3-3-4, FEEDWATER HTR 131-135 LEVEL HIGH, alarms
- H3-3-5, FEEDWATER HTR 131-135 LEVEL HIGH-HIGH, alarms
- A030, FW HTR 132 LVL, alarms

Based on the above conditions, which one of the following is the reactor power one (1) minute later if no operator action is taken?

A. 0% because of an automatic scram.

B. Below 100% but no less than 90%.

C. Unchanged from the previous level.

D. Above 100% but below the scram set point.

Answer: D

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 62 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 62
System ID:	21386
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-260-1-01, EO-1.8b

Answer: D: is correct - Based on the conditions presented, FW HTR 132 trips on high-high level causing a minor loss of FW heating. This failure results in a one (1) degree reduction in the FW temperature returning to the RPV. This reduction in temperature, although small, results in a rise in reactor power. Reactor power will be above 100% but the power change is too small to challenge the flow-biased scram set point.
REFERENCE: DER-NM-2004-5796, and DER-NM-2004-5578, both had the same failure as that presented in the questions. Operators entered N1-SOP-16.1, Feedwater System Failures, and also N1-SOP-1.5, Unplanned Reactor Power Change, and were required to lower reactor recirc flow to remain within the license power limit (100%/1850 mwth).

Distractor: A: is not correct - No scram signal is received. The magnitude of the power change is very small.

Distractor: B: is not correct - Reactor power raises not lowers in response to a loss of FW heating.

Distractor: C: is not correct - The FW heater trips resulting in a small loss of FW heating. Reactor power will

rise above 100%.

REFERENCE:
N1-SOP-1.5, N1-SOP-16.1, DER-NM-2004-5796, DER-NM-2004-5578

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 62 Cross References (table item links)

10CFR55
- 41(b)(5)

Cognitive Level
- 3

DER
- NM-2004-5796, Rev. NA
- NM-2004-5578, Rev. NA

NUREG 1123 KA Catalog Rev. 2
- 259001 K3.12 3.8/3.9 Reactor power

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-001-260-1-01 Rev. na

Question Source
- New

PROC
- N1-SOP-16.1 Rev. NA
- N1-SOP-1.5 Rev. NA

Question Setting
- C1 (License class closed reference)

63 **SYSID: 21387** **Points: 1.00**

The plant is at 100% power. Reactor Building Ventilation will be substituted from System 11 fans in FAST to System 12 Fans in FAST.

Which one of the following is the order of the System 11 fan down shift to SLOW prior to substituting the fans, and the System 12 fan up shift to FAST after substituting fans, to avoid L1-3-4, REACT BLDG/ATM DIFF PRESS, low pressure alarm at -0.05. H2O?

A. For System 11, down shift the supply fan and then the exhaust fan.
For System 12, up shift the supply fan and then the exhaust fan.

B. For System 11, down shift the exhaust fan and then the supply fan.
For System 12, up shift the exhaust fan and then the supply fan.

C. For System 11, down shift the exhaust fan and then the supply fan.
For System 12, up shift the supply fan and then the exhaust fan.

D. For System 11, down shift the supply fan and then the exhaust fan.
For System 12, up shift the exhaust fan and then the supply fan.

Answer: D

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 63 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 63
System ID:	21387
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-288-1-01, EO-1.7b

Answer: D: is correct - To avoid approaching a positive pressure in the reactor building and the low d/p alarm, the exhaust flow is to be maintained higher than the supply flow to ensure the reactor building negative pressure is maintained negative, and the alarm is not received.

Distractor: A: is not correct - Up shift the exhaust fan and then the supply fan, not the supply fan and then the exhaust fan.

Distractor: B: is not correct - Down shift the supply fan and then the exhaust fan, not the exhaust fan and then the supply fan. Up shift the exhaust fan and then the supply fan, not the supply fan and then the exhaust fan.

Distractor: C: is not correct - Down shift the supply fan and then the exhaust fan, not the exhaust fan and then the supply fan.

REFERENCE: N1-OP-10; F.3.1, F.3.2

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 63 Cross References (table item links)

10CFR55

- 41(b)(9)

Cognitive Level

- 2

NUREG 1123 KA Catalog Rev. 2

- 288000 A4.01 3.1/2.9 Start and stop fans

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-288-1-01 Rev. na

Question Source

- New

PROC

- N1-OP-10 Rev. NA

Question Setting

- C1 (License class closed reference)

64 **SYSID: 21388** **Points: 1.00**

Because of entry into N1-EOP-6, Radioactivity Release Control, Control Room EVS is manually placed into service with the following alignment completed one (1) minute ago:

- EMER FAN 11 in RUN
- EMER FAN 12 in OFF
- CONTROL ROOM CIRCULATING FAN 11 in RUN

Which one of the following is (1) the effect of the above configuration and (2) the action to correct it?

A. (1) Control Room Ventilation CHILLER 11 trips on low temperature.
(2) Start Control Room Ventilation CHILLER 12, no reset is required.

B. (1) Control Room Ventilation CHILLER 12 trips on low temperature.
(2) Start Control Room Ventilation CHILLER 11, reset is required first.

C. (1) Increased probability for a loss of control room positive pressure from a single failure.
(2) Change to EMER FAN 12 in RUN and CONTROL ROOM CIRCULATING FAN 12 in RUN.

D. (1) Increased probability for a loss of control room positive pressure from a single failure.
2) Change to EMER FAN 11 in RUN and CONTROL ROOM CIRCULATING FAN 12 in RUN.

Answer: D

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 64 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 64
System ID:	21388
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-288-1-01, EO-1.4b, EO-1.7b

Answer: D. is correct - Per N1-OP-49, H.3.4, D.1.0, E.10.0: To minimize loss of positive pressure in Control Room, lineup Control Room Ventilation fans as follows:
With EMER FAN 11 in AUTO, run CONTROL ROOM CIRCULATING FAN 12 **OR**
With EMER FAN 12 in AUTO, run CONTROL ROOM CIRCULATING FAN 11.
The concern is a single failure (loss of PB16B) will result of a loss of both fans and control room positive pressure degrades. By running EMER FAN 11 (PB16B) and CONTROL ROOM CIRCULATING FAN 12 (PB17B) or EMER FAN 12(PB17) and CONTROL ROOM CIRCULATING FAN 11 (PB16B), this concern is eliminated.

Distractor: A: is not correct - The chillers don't trip based on this alignment. There are other conditions/alignments that result is chiller trips, and in response to a chiller trip, the other/standby chiller is started and then the START push button for each chiller is pressed to reset the chiller compressors.

Distractor: B: is not correct - The chillers don't trip based on this

alignment. There are other conditions/alignments that result is chiller trips, and in response to a chiller trip, the other/standby chiller is started and then the START push button for each chiller is pressed to reset the chiller compressors.

Distractor: C: is not correct - This alignment would introduce the same concern as the lineup presented in the question should a single failure (loss of PB17B) occur and is not an appropriate alignment.

REFERENCE: N1-OP-49, H.3.4, D.1.0, E.10.0

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 64 Cross References (table item links)

10CFR55
- 41(b)(7)

Cognitive Level
- 2

NUREG 1123 KA Catalog Rev. 2
- 290003 A2.01 3.1/3.2 Initiation/reconfiguration

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-001-288-1-03 Rev. na

Question Source
- New

PROC
- N1-OP-49 Rev. NA

Question Setting
- C1 (License class closed reference)

65 **SYSID: 21389** **Points: 1.00**

With the plant at 100% power, a large crack in the outer wall of the steam dryer occurs as indicated on the drawing provided.

Per N1-SOP-1.5, Unplanned Reactor Power Change, which one of the following are the changes in Reactor Power (REACT PWR), Total Reactor Recirc Flow (TOTAL RR FLOW), Reactor Recirc Suction Temp (RR SUCT TEMP), and the Moisture Content in the Steam Leaving the Reactor (CARRYOVER) as a result of this failure?

	REACT PWR	TOT RR FLOW	RR SUCT TEMP	CARRYOVER
A.	Decrease (↓)	Increase (↑)	Increase (↑)	Increase (↑)
B.	Increase (↑)	Decrease (↓)	Decrease (↓)	Decrease (↓)
C.	Decrease (↓)	Increase (↑)	Decrease (↓)	Increase (↑)
D.	Increase (↑)	Decrease (↓)	Increase (↑)	Decrease (↓)

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 65 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 65
System ID:	21389
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-101-1-01, EO-1.7g, EO-1.8b

Determine if any of the following parameter changes occured concurrent with the unexplained power change.

PARAMETERS	Rx Power	Recirc Suct Temp	Rod Line	Total Recirc Flow	Core d/p
Above Top Guide	↓	↑	↓	↑	
Between Top Guide and Core Plate	↓	↑	↓		
Below Core Plate	↓		↓	↑	↓
Minimum Detectable	35-40 MWth	>4°F	2%		

Answer: A: is correct - Per N1-SOP-1.5, Flow Path A: the dryer crack represents an ABOVE TOP GUIDE crack with indications as shown above. Also, per N1-SOP-1.5, 5.3: Failure of Steam Dryer has occurred at Quad Cities. Indications included:
- Lowered Reactor Power due to lowering pressure inside the core shroud.
- Increase in moisture content in steam (carry over).
Moisture Separator failure will have the

same indication as a Dryer failure including steam carry under into the annulus. This will cause abnormal heating of the annulus water raising the recirc pump suction temperature thereby lowering the Net Positive Suction Head (NPSH). Increased moisture carry over will occur due to steam bypassing the Dryer.

Distractor: B: is not correct - REACT PWR, TOTAL RR FLOW, RR SUCT TEMP, and CARRYOVER are all changing in the wrong direction.

Distractor: C: is not correct - RR SUCT TEMP increases, not decreases.

Distractor: D: is not correct - REACT PWR, TOTAL RR FLOW, and CARRYOVER are all changing in the wrong direction.

Question matches the K/A because it evaluates the effect of a malfunction of the reactor vessel internals (steam dryer is a reactor vessel internal component and is malfunctioning because of the crack) on the nuclear boiler instrumentation (reactor power, total recirc flow, and recirc suction temp are all nuclear boiler instruments). Although the moisture carryover is determined by chemistry, it is also used because it is affected as a result of the dryer crack.

REFERENCE:
N1-SOP-1.5, Flowpath A, 5.3, 5.4, Drawing C-35843-C

REFERENCES PROVIDED TO THE CANDIDATE: C-35843-C

Question 65 Cross References (table item links)

10CFR55

- 41(b)(2)

Cognitive Level

- 3

DRW

- C-35843-C Rev. NA

NUREG 1123 KA Catalog Rev. 2

- 290002 K3.07 3.1/3.1 Nuclear boiler instrumentation

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-101-1-01 Rev. na

Question Source

- New

PROC

- N1-SOP-1.5 Rev. NA

Question Setting

- C1 (License class closed reference)

66 **SYSID: 21390** **Points: 1.00**

You have worked all scheduled 12-hour shifts this calendar year as CSO until leaving shift on March 3, 2005 for a Unit 1 refueling outage assignment outside the control room. Since leaving shift, you performed the following 12-hour shifts as the CSO during the refueling outage because of illness:

- March 29, 2005: worked 12 hours as CSO
- March 30, 2005: worked 12 hours as CSO
- April 5, 2005: worked 12 hours as CSO
- April 6, 2005: worked 12 hours as CSO

Which one of the following is correct regarding your license status upon returning to shift on April 10, at the completion of the refueling outage?

A. License was inactive when the April watches were stood requiring a reportable event to the NRC.

B. License requires five 12-hour shifts under instruction to re-activate before assuming shift duties.

C. License is active and only remains active with three additional 12-hour shifts before July 1, 2005

D. License is active and only remains active with one additional 12-hour shift before August 1, 2005.

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 66 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 66
System ID:	21390
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-006-APS-1-01, EO-1.4, EO-1.5

Answer: C: is correct - License is active and only stays active with three additional shifts before 7/1/05. Maintenance of an active license requires that 5-12 hour shifts be stood per calendar quarter. As long as three more 12-hour shift is stood before the end of June (before 7/01/05) the license remains active.

Distractor: A: is not correct - License is still active. The watches stood before being released to the refueling team equate to more than 5-12 hour shifts in a calendar quarter based on the shift schedule.

Distractor: B: is not correct - License is still active. The watches stood before being released to the refueling team equate to more than 5-12 hour shifts in a calendar quarter based on the shift schedule. The candidate could determine that only two of the five required watches were stood in the first calendar quarter if they do not credit the watches before being released to the refueling crew. This is a common misconception of being released from shift responsibilities as termination of an active license requiring reactivation before returning to

shift duties. Additionally, for an inactive license only 4-12 hour watches or 5-8 hour watches are required. If you perform 12 hour shifts then you are required to stand 12-hour watches under instruction to re-activate the license.

Distractor: D: is not correct - Only two watches have been stood in the second calendar quarter (April, May, June) requiring at least three more 12-hour shifts, not just one, be stood by June 30 (before July 1), not August 1. The second calendar quarter ends June 30, not July 31.

REFERENCE: S-ODP-TQS-0101

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 66 Cross References (table item links)

10CFR55
- 41(b)(1)

Cognitive Level
- 2

NUREG 1123 KA Catalog Rev. 2
- G2.1.2 3/4 Knowledge of operator responsibilities during all modes of plant operation

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-006-APS-1-01, Rev. NA

Question Source
- New

PROC
- S-ODP-TQS-0101 Rev. NA

Question Setting
- C1 (License class closed reference)

67 **SYSID: 21391** **Points: 1.00**

Given the following:

- The last watch you stood was day shift on May 1
- You have been on vacation and are preparing to assume the shift as the CSO on day shift on May 9

Per the conduct of operations requirements, which one of the following describes the Control Room Log(s) that MUST BE reviewed BEFORE assuming the shift?

A. Just the night shift Control Room log for May 8. Further back is not required.

B. Just the day and night shift Control Room logs for May 8. Further back is not required and anything less is not appropriate.

C. All Control Room logs back to and including day shift on May 6. Further back is not required and anything less is not appropriate.

D. All Control Room logs back to and including night shift on May 1. Anything less is not appropriate

Answer: D

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 67 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 67
System ID:	21391
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O3-OPS-006-343-3-39, EO-1.5

Answer: D: is correct - Per GAP-OPS-01, 3.4.5, Prior to assuming the shift, the oncoming SM, CRS, STA, CSO, and licensed Reactor Operators shall read applicable logs from the date shift duty was last performed.
Per S-ODP-OPS-0001, 3.3.2.c, Prior to assuming the shift, the oncoming individual shall review pertinent information (such as station logs, records, and special instructions) that have been generated since last on shift as listed on the Shift Turnover Checklist. The oncoming SSS, ASSS and CSO shall read all Control Room Log entries generated since they were last on duty.

Distractor: A: is not correct - Not just the past shift log (12 hours), but all control room logs since the CSO was last on duty. There are other requirements related to the past shift.

Distractor: B: is not correct - Not just the past two shifts logs (24 hours), but all control room logs since the CSO was last on duty. There are other requirements related to the past two shifts.

Distractor: C: is not correct - Not just the past 72 hours, but all control room logs since the CSO was last on duty.

Matches the K/A because one of the shift turnover **practices** is to review the shift log. Knowing how far to look back for specific logs is a key element in this practice.

REFERENCE: GAP-OPS-01, 3.4.5, S-ODP-OPS-001, 3.3.2.c

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 67 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 1

NUREG 1123 KA Catalog Rev. 2
- G2.1.3 3/3.4 Knowledge of shift turnover practices

Level of Difficulty
- Level 2: System operation and response

LP
- O3-OPS-006-343-3-39, Rev. NA

Question Source
- New

PROC
- GAP-OPS-01 Rev. NA
- S-ODP-OPS-001, Rev. NA

Question Setting
- C1 (License class closed reference)

68 **SYSID: 21392** **Points: 1.00**

After responding to an incident involving a contaminated injured worker, the Fire Brigade Leader has requested setup of a Hospital Radiation Emergency Area (REA). Which one of the following is coordinated by the CSO, per EPIP-EPP-04, Personnel Injury Or Illness?

A. Actions to obtain an ambulance and setup a REA at the hospital

B. Actions to reduce the potential for cross-contamination to personnel

C. Actions to notify the NRC of a contaminated injured worker within 8 hours

D. Actions to keep management and hospital staff informed of the workers condition

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 68 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 68
System ID:	21392
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O3-OPS-006-350-3-24, EO-1.4

Answer:	A: is correct - The CSO coordinates the callouts to Oswego 911 and the hospital to setup the REA at the request of the Fire Brigade Leader at the scene.
Distractor:	B: is not correct - The RP Technician is responsible for the radiological aspects of this response including accompanying the worker to the hospital in the ambulance and advising personnel (on site and off site) on actions to avoid cross-contamination.
Distractor:	C: is not correct - This is an event report that is coordinated by the SM.
Distractor:	D: is not correct - This is coordinated both by the Fire Brigade Leader and the CRS/SM as information is gathered regarding the event. The CSO obtains the information, but does not coordinate briefs to management personnel. The hospital staff is briefed by the RP Technician when they arrive at the hospital.
REFERENCE:	EPIP-EPP-04 3.3.2.b, 3.3.2.c, Attachment 1 3.B.1, Attachment 1 3.B.3

REFERENCES PROVIDED TO THE CANDIDATE:
NONE

Question 68 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 1

NUREG 1123 KA Catalog Rev. 2

- G2.1.8 3.8/3.6 Ability to coordinate personnel activities outside the control room

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O3-OPS-006-350-3-24, Rev. NA

Question Source

- New

PROC

- EPIP-EPP-04 Rev. NA

Question Setting

- C1 (License class closed reference)

69 **SYSID: 21393** **Points: 1.00**

Given the following:

- Following completion of a 5-day forced outage, control rod withdrawal for the subsequent startup is ready to start
- Required personnel are stationed and are ready to commence control rod withdrawal
- Before the first control rod is withdrawn, the RWM is declared inoperable

Which one of the following is the correct action to be taken per N1-OP-37, Rod Worth Minimizer (RWM)?

A. Suspend control rod withdrawal until after the RWM fault is repaired, tested, and restored to operable status

B. Bypass the RWM and commence control rod withdrawal since the additional staffing requirement is already met.

C. Bypass the RWM and commence control rod withdrawal after an additional SRO is stationed to verify each control rod movement.

D. Suspend control rod withdrawal until an additional Reactor Engineer is available to verify control rod movement at the completion of each group.

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 69 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 69
System ID:	21393
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-201-1-03, EO-1.5a, EO-1.5b, EO-1.7d

Answer: A. is correct - Since the first 12 control rods of the control rod sequence are not completely withdrawn, control rods shall not be withdrawn. Also since the startup requires additional personnel in the control room for the startup including verification of control rod movements, the TS requirement for additional personnel to verify rod movements is already met and additional staff is not required if 12 control rods had already been withdrawn.

Distractor: B: is not correct - At least 12 control rods must be fully withdrawn to continue control rod withdrawal with the RWM inoperable.

Distractor: C: is not correct - At least 12 control rods must be fully withdrawn to continue control rod withdrawal with the RWM inoperable. Since the startup requires additional personnel in the control room for the startup including verification of control rod movements, the TS requirement for additional personnel to verify rod movements is already met and additional staff is not required if 12 control rods had already been withdrawn.

Distractor: D: is not correct - The reactor engineer is required to verify control rod movement at the completion of each rod group to the process computer printout for all control rod movement with the RWM operable or without the RWM inoperable (after 12 control rods), however, this does not permit continuing with the rod withdrawal without the RWM. At least 12 control rods must be fully withdrawn.

Question matches the K/A because it is an administrative requirement, evaluates pre-startup staffing (for control rod withdrawal), and evaluates whether or not the controls for reactivity changes (control rod withdrawal) can be operated or not.

REFERENCE: N1-OP-37, H.2.2, H.2.3

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 69 Cross References (table item links)

10CFR55
- 41(b)(10)

Cognitive Level
- 2

NUREG 1123 KA Catalog Rev. 2
- G2.2.1 3.7/3.6 Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity

Level of Difficulty
- Level 3: Higher order Knowledge item

LP
- O1-OPS-001-201-1-03, Rev. NA

Question Source
- New

PROC
- N1-OP-37 Rev. NA

Question Setting
- C1 (License class closed reference)

70 **SYSID: 21394** **Points: 1.00**

Which one of the following is a responsibility of the licensed reactor operator in the Control Room during the performance of a fuel shuffle, per N1-FHP-27C, Core Shuffle?

A. Faxes copies of completed fuel movement sheets to the procedure controller

B. Brief the on-coming fuel handling team before they turnover on the refuel floor

C. Observe Source Range Monitor indications when inserting fuel

D. Maintain continuous communications with the Reactor Engineer

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 70 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 70
System ID:	21394
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O3-OPS-006-343-3-36, EO-1.6

Answer: C: is correct

Distractor: A: is not correct - The license reactor operator is the procedure controller. Completed sheets are attached to the control room copy.

Distractor: B: is not correct - This is performed on the refuel floor by the SRO on the refuel bridge. The license operator is the procedure controller for N1-FHP-27C.

Distractor: D: is not correct - The continuous communications are established with the SRO on the refueling bridge. The Reactor Engineer may be on the refueling bridge but is not in control of the evolution and is not the communications interface.

REFERENCE: N1-FHP-27C, N1-FHP-25

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 70 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 1

NUREG 1123 KA Catalog Rev. 2

- G2.2.30 3.5/3.3 Knowledge of RO duties in the control room during fuel handling such as alarms from fuel handling area, communication with fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation

Level of Difficulty

- Level 2: System operation and response

LP

- O3-OPS-006-343-3-36, Rev. NA

Question Source

- New

PROC

- N1-FHP-25 Rev. NA
- N1-FHP-27C Rev. NA

Question Setting

- C1 (License class closed reference)

71 **SYSID: 21395** **Points: 1.00**

Your team is planning a job to be performed in an area classified as a high-radiation area due to a crud trap.

- The dose rate at the component to be worked is **150** mrem/hour.
- Using a long-handled tool reduces the worker's exposure to ¾ the dose rate at the component to be worked.
- The job takes **42** minutes without using the long-handled tool
- The job takes **60** minutes if the long-handled tool is used.
- Installing temporary shielding on the crud trap will lower the dose rate at the component to be worked to **100** mrem/hour.
- Installation and removal of temporary shielding adds **50** mrem of exposure.

Which of the following options satisfies the requirement to perform the job with the least total exposure?

A. Install shielding. Use the tool.

B. Install shielding. Do not use the tool.

C. Do not install shielding. Use the tool.

D. Do not install shielding. Do not use the tool.

Answer: D

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 71 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 71
System ID:	21395
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O3-OPS-006-343-3-40, EO-2.5

Answer: D: is correct - Do not install shielding. Do not use the tool. 150 x .7 = **105 mrem**

Distractor: A: is not correct - Install shielding. Use the tool. 50 + [(100 x 0.75) x 1] = **125 mrem**

Distractor: B: is not correct - Install shielding. Do not use the tool. 50 + (100 x .7) = **120 mrem**

Distractor: C: is not correct - Do not install shielding. Use the tool. 150 x 0.75 x 1.0 = **112.5 mrem**

References: S-RAP-ALA-0102

References Provided: NONE

Modified question from 2004 NRC exam question, Changed values so answer is different.

Question 71 Cross References (table item links)

10CFR55

- 41(b)(12)

Cognitive Level

- 3

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

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O3-OPS-006-343-3-40, Rev. NA

Question Source

- Modified

PROC

- S-RAP-ALA-0102 Rev. NA

Question Setting

- C1 (License class closed reference)

72 **SYSID: 21396** **Points: 1.00**

With the plant operating at 100% power, with a batch discharge in progress.

- H1-4-5, LQ PROCESS RAD MON, in alarm

Which one of the following COMPUTER ALARMS justifies stopping the discharge?

A. E474 RB CLC RMON 11

B. E473 RADWST DIS RMON 11

C. F163 SVC WTR HI RAD RX BLDG

D. F164 SVC WTR HI RAD TURB BLDG

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 72 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 72
System ID:	21396
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O1-OPS-001-273-1-01, EO-1.7e

Answer: B: is correct - Per N1-ARP-H1 H1-4-5: If Liquid Radwaste Discharge monitor alarms, then request the Radwaste Operator to stop the batch discharge (if) in progress.

Distractor: A: is not correct - If Service Water Monitor "HIGH" OR RBCLC "UPSCALE HI-HI", then enter EPIP-EPP-21,Radiation Emergencies, and execute concurrently. If RBCLC monitor alarm, then monitor RBCLC pressure, temperature, make-up tank to determine source of leak and isolate as required (Recirc Pump seal, Radwaste, SDC, RWCU, etc.) and if "UPSCALE HIGH" or "UPSCALE HI-HI", Then notify Chemistry to sample RBCLC as required to locate source of activity.

Distractor: C: is not correct - If Service Water Monitor "HIGH" OR RBCLC "UPSCALE HI-HI", then enter EPIP-EPP-21,Radiation Emergencies, and execute concurrently. If Service Water monitor alarm, then request Chemistry to sample discharge canal and Service Water to locate source of activity.

Distractor: D: is not correct - If Service Water Monitor "HIGH" OR RBCLC "UPSCALE HI-HI", then enter EPIP-EPP-21,Radiation Emergencies, and execute concurrently. If Service Water

monitor alarm, then request Chemistry to sample discharge canal and Service Water to locate source of activity.

REFERENCE: N1-ARP-H1, H1-4-5.

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 72 Cross References (table item links)

10CFR55

- 41(b)(11)
- 41(b)(8)

Cognitive Level

- 1

NUREG 1123 KA Catalog Rev. 2

- G2.3.11 2.7/3.2 Ability to control radiation releases

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O1-OPS-001-273-1-01, REV. NA

Question Source

- New

PROC

- N1-ARP-H1 Rev. NA

Question Setting

- C1 (License class closed reference)

73 **SYSID: 21397** **Points: 1.00**

You are on shift as the Chief Shift Operator when the I&C Supervisor call reporting the disappearance of one of the I&C Techs dispatched to perform a calibration activity in the Unit 1 Reactor Building. Which one of the following is one of your responsibilities per EPIP-EPP-03, Search and Rescue for this event?

A. Designate a licensed reactor operator to act as the Search and Rescue Team Leader

B. Notify security and request the run of an ACAD report to determine if the tech is on site

C. Dispatch Nuclear Security with a radio, to the search and rescue command post

D. Gather information from station staff and security to determine potential locations

Answer: B

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 73 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 73
System ID:	21397
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O3-OPS-006-350-3-23, EO-1.4

Answer: B: is correct - Per EPIP-EPP-03, Search and Rescue, the CSO executes the Attachment 1 Checklist. Item #9 on the checklist is "Security notified / ACAD run requested".

Distractor: A: is not correct - The Shift Manager designates the Search and Rescue Team Leader.

Distractor: C: is not correct - Direct the Unit 2 to page for the missing I&C tech from Unit 2 Control Room

Distractor: D: is not correct - This is a responsibility of the Search and Rescue Team Leader.

REFERENCE: EPIP-EPP-3, Search and Rescue

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 73 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 1

NUREG 1123 KA Catalog Rev. 2

- G2.4.12 3.4/3.9 Knowledge of general operating crew responsibilities during emergency operations

Level of Difficulty

- Level 2: System operation and response

LP

- O3-OPS-006-350-3-23, Rev. NA

Question Source

- New

PROC

- EPIP-EPP-03 Rev. NA

Question Setting

- C1 (License class closed reference)

74 **SYSID: 21398** **Points: 1.00**

Given the following conditions:

- The available Fire Brigade consists of the minimum number of members allowed by station procedures
- An operator responds to an electrical fire as the Control Room Liaison for the fire brigade
- The operator brings a Class B/C fire extinguisher to the scene
- The other fire brigade members rig a fire hose with a solid-stream nozzle to the scene
- All required fire brigade members are at the scene
- The Fire Chief assumes the role as the Incident Commander and directs the fire fighting activities

Which one of the following is the number of fire brigade members (not counting the Control Room Liaison) being directed by the Incident Commander and the fire fighting equipment to extinguish this fire?

A. Four (4) fire brigade members. Do not use the fire hose. Put the fire out with the Class B/C fire extinguisher

B. Four (4) fire brigade members. Use the fire hose first. If it does not put out the fire, use the Class B/C fire extinguisher.

C. Five (5) fire brigade members. Do not use the fire hose. Put the fire out with the Class B/C fire extinguisher.

D. Five (5) fire brigade members. Get a Class D fire extinguisher brought to the scene and then use the Class D fire extinguisher.

Answer: A

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 74 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 74
System ID:	21398
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	O3-OPS-006-350-3-42, EO-1.4

Answer: A: is correct - Fire brigade consists of five personnel including the Fire Chief. With the Fire Chief as the Incident Commander, only 4 fire brigade members remain to be directed at the scene. An electrical fire is a Class C fire. Use a Class B/C fire extinguisher such as CO2. Do not use water on electrical.

Distractor: B: is not correct - Use the Class B/C extinguisher first.

Distractor: C: is not correct - Only 4 fire brigade members remain after the Fire Chief assumes the role as the Incident Commander.

Distractor: D: is not correct - Only 4 fire brigade members remain after the Fire Chief assumes the role as the Incident Commander. Class C is the correct extinguisher.

REFERENCE: NIP-FPP-01, EPIP-EPP-28

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 74 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 1

NUREG 1123 KA Catalog Rev. 2

- G2.4.26 2.9/3.3 Knowledge of facility protection requirements including fire brigade and portable fire fighting equipment usage

Level of Difficulty

- Level 3: Higher order Knowledge item

LP

- O3-OPS-006-350-3-42, Rev. NA

Question Source

- New

PROC

- EPIP-EPP-28 Rev. NA
- NIP-FPP-01 Rev. NA

Question Setting

- C1 (License class closed reference)

75 **SYSID: 21399** **Points: 1.00**

The plant is operating at power, with the following:

- Shift Manager has declared an UNUSUAL EVENT due to Seismic Event
- The extra Reactor Operator normally assigned to the WEC is dispatched to the Reactor Building to perform damage control actions
- While in the Reactor Building, a station announcement is made upgrading the event to an ALERT

Which one of the following actions is required by the extra RO, per EPIP-EPP-22, Damage Control?

A. Immediately report to the WEC and wait to be assigned to a repair team

B. Immediately report to the OSC and wait to be assigned to a repair team

C. Contact the Shift Manager from the Reactor Building for directions

D. Contact the OSC Coordinator from the Reactor Building for directions

Answer: C

Associated objective(s):

Development Area (FIO)

EXAMINATION ANSWER KEY

UNIT 1 REACTOR OPERATOR 2005 EXAM

Question 75 Details

Question Type:	Multiple Choice
Topic:	RETAKE RO 75
System ID:	21399
User ID:	
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	0
Point Value:	1.00
Cross Reference:	
User Text:	
User Number 1:	0.00
User Number 2:	0.00

Comment:

Answer: C. is correct - Per EPIP-EPP-22, if directly involved in damage control, contact the SM and request directions.

Distractor: A: is not correct - Per EPIP-EPP-22, if directly involved in damage control, contact the SM and request directions. If you were directed to leave the plant, you would return to the Control Room.

Distractor: B: is not correct - Per EPIP-EPP-22 if directly involved in damage control, contact SM and request directions. If you were directed to leave the plant, you would return to the Control Room. You would not be sent to the OSC until is is staffed.

Distractor: D: is not correct - Per EPIP-EPP-22, if directly involved in damage control, contact SM and request directions. Once the OSC is staffed, you would be sent to report to the DCT Coordinator.

REFERENCE: EPIP-EPP-22 Section 3.9

REFERENCES PROVIDED TO THE CANDIDATE: NONE

Question 75 Cross References (table item links)

10CFR55

- 41(b)(10)

Cognitive Level

- 1

NUREG 1123 KA Catalog Rev. 2

- G2.4.29 2.6/4 Knowledge of the emergency plan

Level of Difficulty

- Level 2: System operation and response

LP

- O3-OPS-006-350-3-38, Rev. NA

Question Source

- New

PROC

- EPIP-EPP-22 Rev. NA

Question Setting

- C1 (License class closed reference)