U.S. Nuclear Regulatory Commission 2012 SRO Written Exam (Quad Cities)

Test Answer Key Final

76 **ID: QDC.ILT.17133** **Points: 1.00**

Unit 1 is at full power when annunciator 901-7 D-2, TURB GEN BRG HI/HI-HI VIBRATION, alarms.
The NSO reports that the two highest Turbine bearing vibration readings reached were 11.5 mils and that current readings are now at 11.0 mils and lowering.

(1) Which of the following describes the status of the Main Turbine?

(2) What action is the Unit Supervisor required to direct NEXT?

A. (1) A Turbine trip has automatically occurred
(2) Enter QOA 5600-04, Loss of Turbine Generator, and verify Turbine speed is decreasing

B. (1) A Turbine trip did NOT automatically occur
(2) Enter QCOA 5600-01, Main Turbine High Vibration, and trip the Turbine

C. (1) A Turbine trip did NOT automatically occur
(2) Enter QCOA 5600-01, Main Turbine High Vibration, and throttle open the MO 1-4901 TURB VAC BKR

D. (1) A Turbine trip did NOT automatically occur
(2) Enter QCOA 5600-01, Main Turbine High Vibration, and Reduce Turbine load per QCGP 3-1 to reduce vibration levels to < 7 mils

Answer: B

Answer Explanation:

QCAN 901(2)-7 D-2, states the that if the High Vibration Turbine trip is enabled then the Turbine trips at 10 mils. Currently, no procedural guidance exists to enable the high vibration trip, and it is normally not enabled.
With vibration readings >10 mils and decreasing, the US will direct QCOA 5600-01 entered and the Turbine tripped. This is not an immediate operator action and requires the US permission/direction to execute.

Distractor 1 is incorrect: Plausible because this would be the correct answer if the High Vibration trip was enabled.
Distractor 2 is incorrect: Plausible because this action would be correct if vibration readings exceeds 12 mils.
Distractor 3 is incorrect: Plausible because this would be the correct answer if vibration readings were less than 10 mils.

Reference: QCAN 901(2)-7 D-2 rev 7, QCOA 5600-01 rev 11
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): SRO
Tier: 1 **Group:** 1

K/A: 295005 Main Turbine Trip
2.1.20 Ability to interpret and execute procedure steps. (RO=4.6 / SRO=4.6)

SRO Justification: 10 CFR 55.43(b)(5)

Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed

Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Question Source: New
Question History: N/A

Comments:

Associated objective(s):

SR-5600-K20 (Freq: LIC=B)
Given a Main Turbine and Auxiliary Systems operating mode and various plant conditions, EVALUATE the following Main Turbine and Auxiliary Systems indications/responses and DETERMINE if the indication/ response is expected and normal.

a. Main turbine speed
b. Turbine steam flow
c. Turbine pressures
  (1) Steam chest
  (2) Throttle
  (3) First stage
  (4) HP exhaust
  (5) LP inlet
d. Turbine eccentricity and vibrations
e. Turbine expansion and metal temperatures
f. Thrust bearing wear detector
g. Turning gear running/engaged indications and motor current
h. Shaft sealing system
  (1) Seal steam pressure
  (2) Condenser level
  (3) Valve positions
  (4) Gland exhaust pressure
i. Hood spray
  (1) Exhaust hood temperature
  (2) Hood spray valves position
j. Turbine oil
  (1) Turbine oil reservoir level
  (2) Filter pump discharge pressure
  (3) Filter system flow
  (4) Filter differential pressure
  (5) Turbine oil pressures:
    (a) bearing header
    (b) operating oil
    (c) main pump suction
    (d) main discharge
    (e) booster
    (f) booster nozzle
  (6) Oil cooler outlet temperature
  (7) Lift pump discharge pressure
  (8) MSP / TGOP / EBOP run/pressure indications
  (9) Vapor extractor pressure light (e) booster
    (f) booster nozzle
  (6) Oil cooler outlet temperature
  (7) Lift pump discharge pressure
  (8) MSP / TGOP / EBOP run/pressure indications
  (9) Vapor extractor pressure light

295005.2.1.20 Ability to interpret and execute procedure steps. (RO=4.6 / SRO=4.6)

77	ID: QDC.ILT.17134	Points: 1.00

The following conditions exist on Unit 2:

- Reactor Mode Switch is in REFUEL.
- Mode 5 was entered 8 DAYS ago.
- Fuel moves are in progress within the Reactor Pressure Vessel (RPV)
- Control Rod Drive (CRD) removal from under the vessel is in progress. This activity has been screened as an Operation with the Potential to Drain the Reactor Vessel (OPDRV).
- The 'A' Control Room HVAC Train is running

If a loss of RHR Service Water to the condensing unit of the 'B' Control Room HVAC Train occurs, which of the following is the correct action, if any, to be taken in regards to LCO 3.7.5, Control Room Emergency Ventilation Air Conditioning System?

A. NO action required. LCO 3.7.5 is met with the remaining OPERABLE equipment.

B. NO action required. Control Room HVAC condensing unit cooling water automatically swaps to Service Water.

C. Immediately initiate action to suspend the CRD removal work.

D. Initiate action to suspend the CRD removal work within ONE hour.

Answer: C

Answer Explanation:

The Control Room Emergency Ventilation AC system consists of only the 'B' Train of CRHVAC. RHRSW is required in Modes 4 and 5 in order for CRHVAC to be operable. With Unit 2 in Mode 5 with OPDRVs in progress, actions to suspend the under vessel work must be initiated immediately IAW LCO 3.7.5 Condition C.

Distractor 1 is incorrect: Plausible if the candidate assumes the 'A' Train of CRHVAC will satisfy the LCO requirements.
Distractor 2 is incorrect: Plausible if the candidate assumes the cooling water will automatically swap.
Distractor 3 is incorrect: Plausible if the candidate assumes the actions of LCO 3.7.5 Condition C are required within 1 hour, as many of the shutdown LCO actions are.

Reference: Tech Spec Bases B 3.7.5 Control Room Emergency Ventilation AC System rev 31, Tech Spec LCO 3.7.5 Control Room Emergency Ventilation AC System ammendment 245/240

Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): SRO
Tier: 1 **Group:** 1

K/A: 295018 Partial or Complete Loss of Component Cooling Water

2.2.37 Ability to determine operability and/or availability of safety related equipment. (RO=3.6 / SRO=4.6)

Question Source: New
Question History: N/A

SRO Justification: 10 CFR 55.43(b)(2)

Can question be answered *solely* by knowing ≤ 1 hour TS/TRM Action?
NO
Can question be answered *solely* by knowing the LCO/TRM information listed “above-the-line?”
NO
Can question be answered *solely* by knowing the TS Safety Limits?
NO

Does the question involve one or more of the following for TS, TRM, or ODCM?

- Knowledge of TS bases that is required to analyze TS required actions and terminology

The question requires the candidate to use TS bases knowledge to determine the operability of the CRHVAC system.

Comments:

Associated objective(s):

S-5752-K33 (Freq: LIC=I)
DISCUSS the bases for Control Room Ventilation System Tech Spec LCO, related safety limits and LSSS's

295018.2.2.37 Ability to determine operability and/or availability of safety related equipment. (CFR 41.7 / 43.5 / 45.12) (RO=3.6 / SRO=4.6)

78 **ID: QDC.ILT.17105** **Points: 1.00**

Unit 1 is at rated power.

Unit 2 was in Mode 4 with the 'X-1 Hatch' removed and both Reactor Recirc Pumps off when a loss of Shutdown Cooling (SDC) occurs.

Which ONE of the following completes both statements?

The **minimum** required Reactor Water Level to support Natural Circulation is _(1)_ inches.

A 'Site Area Emergency' declaration is **first** required when RPV water level drops below _(2) inches.

MS8 Loss of RCS/RPV inventory affecting core decay heat removal capability. **4**

EAL Threshold Values:

NOTE: The Emergency Director should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

1. **Without** Primary or Secondary CONTAINMENT CLOSURE established:
   a. RPV level < **- 65 in**.
   OR
   b. RPV level unknown for > **30 minutes** with a loss of RPV inventory per Table M5 indications.

OR

2. **With** Primary or Secondary CONTAINMENT CLOSURE established:
   a. RPV level < **- 142 in.** (TAF).
   OR
   b. RPV level unknown for > **30 minutes** with a loss of RPV inventory as evidenced by either of the following:
      - Per Table M5 indications.
      - Erratic Source Range Monitor indication.

A. (1) 90
(2) -65

B. (1) 90
(2) -142

C. (1) 100
(2) -65

D. (1) 100
(2) -142

Answer: B

Answer Explanation:

The minimum level required to support natural circulation is 90 inches. QCOA 1000-02, Loss of Shutdown Cooling, directs RPV water level to be maintained in a band of 90-100 inches.
The EAL threshold value for MS8 is -142 inches. The X-1 Hatch disables primary containment, but secondary containment is still established (Unit 1 at full power requires secondary containment established)

Distractor 1 is incorrect: Plausible because this would be the correct answer if neither Primary or Secondary containment were established.
Distractor 2 is incorrect: Combination of distractors 1 & 3.
Distractor 3 is incorrect: Plausible because RPV water level must be raised to a band of 90-100 inches, however, the question is asking the minimum level required to support natural circulation.

Reference: LIC-0201 rev 14, QCOA 1000-02 rev 17, EP-AA-1006 rev 31
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): SRO
Tier: 1 **Group:** 1

K/A: 295021.AA2.03 Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: (CFR 41.10)
Reactor water level (RO=3.5 / SRO=3.5)

SRO Justification: Unique to SRO position (reference facility objective)

Question Source: New
Question History: Modeled from a question from Brunswick 2010 ILT NRC Exam

Comments:

Associated objective(s):

295021.AA2.03 Reactor water level (RO=3.5 / SRO=3.5)

S-1000-K70 (Freq: LIC=B ILT=NA) Given a RHR System operating mode and various plant conditions and a copy of EP-AA-111 and EP-AA-1006, CLASSIFY the event/abnormal condition including correct EALs and PARs in accordance with EP-AA-111 and EP-AA-1006.

79 **ID: QDC.ILT.17145** **Points: 1.00**

Unit 2 is at 100% power when a transient occurs, resulting in a HIGH reactor pressure condition.

Which of the following conditions, if any, will require the Shift Manager to declare an ALERT due to meeting the threshold values of MA3? If neither condition requires a declaration, why?

Condition 1: RPV pressure reaches 1080 psig and reactor shutdown is achieved by manual scram insertion.

Condition 2: RPV pressure reaches 1257 psig and reactor shutdown is achieved by automatic ARI initiation.

(Note: Consider each condition SEPARATELY)

MA3 Failure of the Reactor Protection System to complete or initiate an automatic reactor scram once a Reactor Protection System setpoint has been exceeded. $\boxed{1}\boxed{2}$

EAL Threshold Values:

1. A Reactor Protection System setpoint was exceeded.

   AND

2 Automatic scram did not reduce reactor power to < **IRM Range 7 and lowering**.

A. Condition 1 and 2

B. Condition 1 only

C. NEITHER because MA3 is NOT applicable for the initial operating condition.

D. NEITHER because a complete reactor shutdown was achieved in both conditions.

Answer: A

Answer Explanation:

The candidate must determine:
1) if reactor pressure is above the RPS scram setpoint.
2) if ARI and manual scram insertion are normal or alternate means of rod insertion.
3) if MA3 is applicable if the reactor is shutdown and in Mode 3.

The UFSAR analytical value for RPS scram on high RPV pressure is 1060 psig (actual is 1024 psig). Therefore, anything less than 1024 psig does not meet threshold value one (1).

The second condition of this EAL indicates a failure of the automatic RPS scram function to rapidly insert a sufficient number of control rods to achieve reactor shutdown.

Even if the first subsequent manual scram signal inserts all control rods to the full-in position immediately after the initial failure of the automatic scram, the lowest level of classification that must be declared is an Alert.

The Alternate Rod Insertion (ARI) system provides an automatic and alternate method of completing the scram function. This backup, however, inserts control rods at a much slower rate than the automatic RPS scram function. For the purpose of emergency classification at the Alert level, reactor shutdown achieved by ARI initiation does not constitute a successful RPS automatic scram.

Since the ATWS condition (for Condition 2) was initially present with the plant in Mode 1, MA3 is applicable (even if ARI places the plant in Mode 3).

Distractor 1 is incorrect: Plausible if candidate does not recognize that reactor shutdown from manual scram insertion after exceeding an automatic scram setpoint meets the MA3 classification threshold.
Distractor 2 is incorrect: Plausible if candidate incorrectly applies the applicability of MA3 to the initial condition of the plant.
Distractor 3 is incorrect: Plausible if candidate assumes that declaring an alert is not required because the reactor was shutdown in both conditions.

Reference: EP-AA-1006 rev 31
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): SRO
Tier: 1 **Group:** 1

K/A: 295037.EA2.06 Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN :
Reactor Pressure (RO=4.0 / SRO=4.1)

Question Source: Modified from 2011 Quad Cities ILT NRC Exam
Question History: N/A

SRO Justification: Unique to SRO position (reference facility objective)

Comments: None

Associated objective(s):

295037.EA2.06 Reactor pressure (RO=4.0 / SRO=4.1)

S-0303-K70 (Freq: LIC=B ILT=NA) Given an ATWS/ARI System operating mode and various plant conditions and a copy of EP-AA-111 and EP-AA-1006, CLASSIFY the event/abnormal condition including correct EALs and PARs in accordance with EP-AA-111 and EP-AA-1006.

80	ID: QDC.ILT.17100	Points: 1.00

(A reference is provided for this question)

During an accident on Unit 1 the following plant conditions exist:

- RPV pressure 600 psig
- RPV water level 30 inches
- Drywell pressure 17 psig
- Torus water level 12 feet
- Torus pressure 16 psig
- Drywell sprays are unavailable

Which ONE of the following is required based on the above conditions?

A. Anticipate Blowdown and rapidly depressurize using the Turbine Bypass Valves IAW QGA 100, RPV Control

B. Establish a reactor cooldown using Turbine Bypass Valves at a rate < 100 ºF/hr IAW QGA 100, RPV Control

C. Enter QGA 500-1, RPV Blowdown, and emergency depressurize using the ADS valves

D. Enter QGA 500-1, RPV Blowdown, and emergency depressurize using Turbine Bypass Valves or other Emergency Depressurization Systems

Answer: C

Answer Explanation:

Torus level and pressure conditions are in violation of the Pressure Suppression Pressure limit (Detail L, QGA 200) and an Emergency Depressurization is required. ADS valves are used to blowdown the reactor with Torus level above 5 feet.

Distractor 1 is incorrect: Plausible because anticipating a blowdown on Torus level can be used until conditions requiring a rapid blowdown using ADS valves are met. The conditions of the stem require a blowdown using ADS valves.
Distractor 2 is incorrect: Plausible if candidate rules out anticipating blowdown due to RPV level and does not recognize that Torus conditions require a blowdown.
Distractor 3 is incorrect: Plausible because Torus water level is low, but not low enough to prohibit the use fo ADS valves and use other Emergency depressurization methods instead.

Reference: QGA 200 rev 9 Primary Containment Control, QGA 100 rev 9 RPV Control, QGA 500-1 rev 13 RPV Blowdown
Reference provided during examination: QGA 200 rev 9 with entry conditions/information removed

Cognitive level: High

Level (RO/SRO): SRO
Tier: 1 **Group:** 1

K/A: 295030.EA2.01 Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR 41.10)
Suppression pool level (RO=4.1 / SRO=4.2)

SRO Justification: 10 CFR 55.43(b)(5) Requires knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures

Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Question Source: Modified from DAEC 2009 ILT NRC Exam
Question History: N/A

Comments:

Associated objective(s):

295030.EA2.01 Suppression pool level (RO=4.1 / SRO=4.2)

S-0001-K24 (Freq: LIC=B)
Given QGA 200, 'Primary Containment Control' and QGA 200-5, 'Hydrogen Control', and various conditions, EVALUATE the conditions and DESCRIBE how to proceed through the flowcharts including transitions within QGA 200 or 200-5, to other QGA procedures or to normal operating procedures.

81 **ID: QDC.ILT.15685** **Points: 1.00**

Unit 1 was at full power when there was a loss of offsite power and reactor scram. Both Diesel Generators autostarted and loaded to their respective busses.

As the plant is being stabilized, a fire is reported on Turbine Building 595' level near the 1/2B Instrument Air Compressor.

The EO at the U-1 EDG reports that the Room Vent Fan is NOT running, isolation dampers are closed and the green light associated with the Vent Fan control switch on the 2251-37 panel is LIT.

Assuming damaged fire detection cabling is locking out the Unit 1 EDG Vent Fan:

(1) What procedure must be entered?

(2) What actions must the Unit Supervisor direct to be taken?

A. (1) Enter QCOA 6600-08, UNIT 1(2) DIESEL GENERATOR ROOM VENT FAN FAILURE

(2) Direct the EO to swap the Vent Fan power supply to the Alternate Feed, start the Vent Fan and verify the Isolation Dampers OPEN.

B. (1) Enter QCOP 6600-11, DIESEL GENERATOR LOCAL OPERATION

(2) Direct the EO to perform a emergency shutdown of the U-1 EDG.

C. (1) Enter QCOA 6600-08, UNIT 1(2) DIESEL GENERATOR ROOM VENT FAN FAILURE

(2) Direct the EO to place the D.G. 1 VENT FAN FIRE PROTECTION BYPASS SWITCH in BYPASS and verify that the Vent Fan starts and Isolation Dampers OPEN.

D. (1) Enter QCOA 6600-08, UNIT 1(2) DIESEL GENERATOR ROOM VENT FAN FAILURE

(2) Direct the EO to depress the RESET pushbutton at Panel 2212-47, CO_2 Electrical Control Cabinet located near the Cardox Tank.

Answer: C

Answer Explanation:

Damage to the fire detection cabling causes an auxiliary relay (AR) to energize, opening contacts and de-energizing SO 1-5799-553. This closes the room isolation dampers and stops the DG Vent Fan. SO 1-5799-89 is also de-energized closing the normal ventilation duct isolation dampers. The EDG is needed since there was LOOP, therefore to maintain room ventilation, the 'D.G. 1 VENT FAN FIRE PROTECTION BYPASS SWITCH' is placed in BYPASS per QCOA 6600-08.

Distractor 1 is incorrect: Swapping to the alternate power supply is only necessary if the normal is lost. Since the green light on the 2251-97 panel is lit, this is not the case. Even if the Vent fan is placed on the alternate power supply, the fan will still remain tripped due to the fire signal.

Distractor 2 is incorrect: The EDG is needed for emergency use, therefore it should not be tripped.

Distractor 3 is incorrect: The EO should not be sent to the Fire Protection panel due to the fire location. Even if the RESET pushbutton is depressed, the cabling is damaged and the fan will remain tripped.

Reference: QCOA 6600-08 rev 10
Reference provided during examination: None.

Cognitive level: High

Level (RO/SRO): SRO
Tier: 1 **Group:** 1

K/A: 600000.AA2.02 Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE:
Damper position (RO=2.8 / SRO=2.9)

SRO Justification: 10 CFR 55.43(b)(5) Requires knowledge of diagnostic steps and decision points in the abnormal procedures that involve transitions to event specific sub-procedures

Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? NO

Can the question be answered solely by knowing immediate operator actions? NO

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? NO

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? NO

Question Source: Quad Cities ILT Exam Bank
Question History: Quad Cities 2009 ILT Cert Exam

Comments:

Associated objective(s):

600000.AA2.02 Damper position (RO=2.8 / SRO=2.9)

SRN-6600-K26 (Freq: LIC=B NF=B) EVALUATE given key Emergency Diesel Generator parameter indications and/or responses depicting a system specific abnormality/failure and DETERMINE a course of action to correct or mitigate the following abnormal condition(s):

a. Loss of diesel cooling water pump
b. Loss of fuel oil transfer pump (use of portable pump)
c. CO2 fire protection system initiation from false signal
d. Fuel oil spill >15 gallons

82 **ID: QDC.ILT.17146** **Points: 1.00**

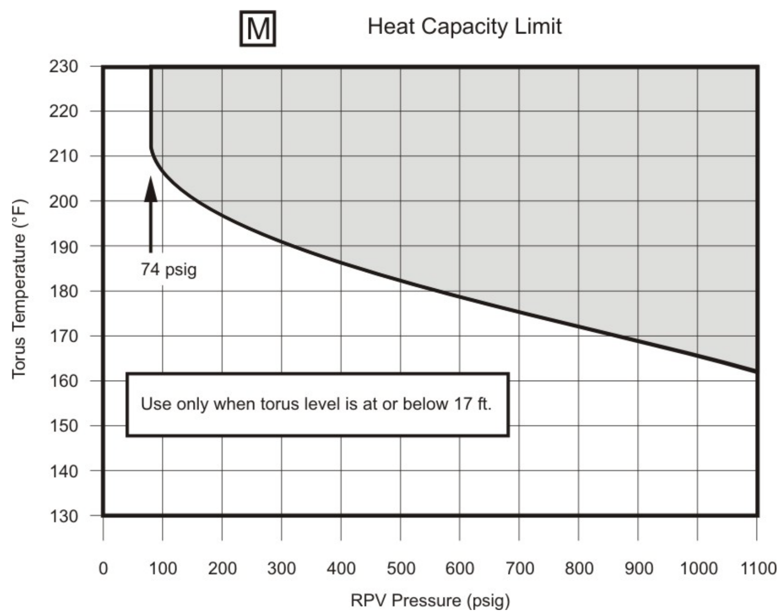
Unit 2 was at rated power when an accident occurred resulting in the following plant conditions:

- Torus water level is 12.0 feet and steady
- Torus water temperature is 180°F and steady
- Reactor pressure is 400 psig and steady

Complete the following statements:

Given the above plant conditions, the Heat Capacity Temperature Limit (HCTL) __(1)__ exceeded.

Per the EOP basis, maintaining plant conditions within the HCTL will __(2)__.



A. (1) IS
(2) prevent ECCS pump damage from inadequate net positive suction head

B. (1) IS NOT
(2) prevent ECCS pump damage from inadequate net positive suction head

C. (1) IS
(2) ensure Torus pressure remains below the Primary Containment Pressure Limit (PCPL) during a RPV blowdown

D. (1) IS NOT
(2) ensure Torus pressure remains below the Primary Containment Pressure Limit (PCPL) during a RPV blowdown

Answer: D

Answer Explanation:

The HCTL is the highest torus temperature from which a RPV blowdown will not raise (a) torus temperature above the torus design temperature and (b) torus pressure above the PCPL, while the rate of energy transfer from the RPV to the primary containment is greater than the capacity of the containment vent.

Distractor 1 is incorrect: Plausible because pump damage is a concern when suppression pool temperature is high (displayed as a caution in the EOP torus temperature leg). However, a separate graph is used to determine margin to pump damage from inadequate NPSH.
Distractor 2 is incorrect: Combination of distractor 1 and 2.
Distractor 3 is incorrect: Plausible if candidate determines that operation must be above the line, similar to the BFPL curves for the RPV.

Reference: L-QGADET Rev 8
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): SRO
Tier: 1 **Group:** 1

K/A: 295026 Suppression Pool High Water Temperature
2.4.18 Knowledge of the specific bases for EOPs. (RO=3.3/ SRO=4.0)

Question Source: Modified from Quad Cities 2011 ILT NRC Exam
Question History: N/A

SRO Justification: Candidate must perform actions that are unique to the SRO position (reference facility objective).

Comments: None

Associated objective(s):

S-0001-K12a (Freq: LIC=B)
EVALUATE given system/plant parameters and the following QGA curves/tables, DETERMINE if any QGA related limits have been exceeded:
a. QGA Detail A, RPV Water Level Instruments
   1. Figure B, RPV Saturation Curve
   2. Table C, RPV Level Instrument Criteria
b. QGA Figure D, Primary Containment Pressure Limit
c. QGA Table J, Minimum Steam Cooling Pressure
d. QGA Figure K, Drywell Spray Initiation Limit
e. QGA Figure L, Pressure Suppression Pressure
f. QGA Figure M, Heat Capacity Limit
g. QGA Table S, Reactor Building Area Temperatures
h. QGA Table T, Reactor Building Area Radiation Levels
i. QGA Table U, Reactor Building Area Water Levels

295026.2.4.18 Knowledge of the specific bases for EOPs. (RO=3.3 / SRO=4.0)

83 **ID: QDC.ILT.17149** **Points: 1.00**

An NSO reports that Drywell Average Air Temperature is 145ºF.

This temperature __(1)__ exceed the Technical Specification Limit,
and is based on preventing __(2)__ during a DBA LOCA.

A. (1) DOES
(2) exceeding 150ºF in the Torus and the potential degradation of the primary containment structure under accident loads

B. (1) DOES
(2) exceeding 281ºF in the Drywell and the potential degradation of the primary containment structure under accident loads

C. (1) DOES NOT
(2) exceeding 150ºF in the Torus and the potential degradation of the primary containment structure under accident loads

D. (1) DOES NOT
(2) exceeding 281ºF in the Drywell and the potential degradation of the primary containment structure under accident loads

Answer: D

Answer Explanation:

Per Tech Spec Bases 3.6.1.5, the LCO limit is 150 F based on meeting the initial conditions of the safety analysis, not to exceed peak LOCA temperature of 281 F.

Distractor 1 is incorrect: Combination of distractors 2 and 3.
Distractor 2 is incorrect: The Tech Spec limit for Drywell Temperature is 150ºF.
Distractor 3 is incorrect: The basis for the Tech Spec limit on Drywell temperature is based on protecting the Drywell, not the Torus.

Reference: Tech Spec Bases 3.6.1.5 rev 0
Reference provided during examination: None

Cognitive level: Memory

Level (RO/SRO): SRO
Tier: 1 **Group:** 2

K/A: 295012 High Drywell Temperature
2.2.22 Knowledge of limiting conditions for operations and safety limits. (RO=4.0 / SRO=4.7)

SRO Justification: 10 CFR 55.43(b)(2)

Can question be answered *solely* by knowing ≤ 1 hour TS/TRM Action?
NO
Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?"
NO

Can question be answered *solely* by knowing the TS Safety Limits?
NO

Does the question involve one or more of the following for TS, TRM, or ODCM?

- Knowledge of TS bases that is required to analyze TS required actions and terminology

Question Source: Quad Cities ILT Exam Bank
Question History: N/A

Comments:

Associated objective(s):

S-1601-K33 (Freq: LIC=I)
DISCUSS the bases for Containment Systems Tech Spec LCO's.

295012.2.2.22 Knowledge of limiting conditions for operations and safety limits. (RO=3.4 / SRO=4.1)

84 **ID: QDC.ILT.17081** **Points: 1.00**

Unit 1 is in Mode 5 with the core fully loaded.

- ONE (1) control rod is fully withdrawn
- All other control rods are fully inserted
- Both CRD pumps have tripped
- CRD Charging Water header pressure is 0 psig
- NO CRD Accumulator Low Pressure alarms are present

Based on the given conditions, what action is the Unit Supervisor required to direct?

A. Place both CRD Pump Suction Filters in operation IAW QCOP 0300-12, CRD System Suction Filter Operation

B. Fully insert the INOPERABLE control rod within 3 hours AND disarm the associated CRD IAW LCO 3.1.3 Control Rod Operability, Condition C

C. Immediately initiate action to fully insert the withdrawn control rod IAW QCOA 0300-01, Control Rod Drive Pump Failure

D. Immediately suspend CORE ALTERATIONS except for control rod insertion and fuel assembly removal IAW LCO 3.1.1 Shutdown Margin, Condition E

Answer: C

Answer Explanation:

In Modes 3, 4, or 5 QCOA 0300-01 requires any withdrawn control rod to be inserted anytime there is no CRD pump running.
The absence of any CRD Accumulator alarms indicates that accumulator pressure is greater than 940 psig.

Distractor 1 is incorrect: Plausible because this is an action in the Low CRD Drive Water Pressure procedure, however, the action is based on CRD accumulator pressure.
Distractor 2 is incorrect: Plausible because this would be the correct answer if the Reactor was in Mode 1 or 2.
Distractor 3 is incorrect: Plausible if candidate believes that Shutdown Margin is not met. SDM is assured with one or less control rods fully withdrawn from the core with fuel in the RPV. This would be the correct answer if there were more than one rod withdrawn while in Mode 5.

Reference: QCOA 0300-01 rev 16
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): SRO
Tier: 1 **Group:** 2

K/A: 295022 Loss of Control Rod Drive Pumps
295022.2.4.11 Knowledge of abnormal condition procedures. (RO=3.4 / SRO=3.6)

SRO Justification: 10 CFR 55.43(b)(5) Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure to mitigate, recover, or with which to proceed

Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Question Source: New
Question History: N/A

Comments:

Associated objective(s):

295022.2.4.11 Knowledge of abnormal condition procedures. (RO=3.4 / SRO=3.6)

SR-0302-K22 (Freq: LIC=B)
Given a Control Rod Drive Hydraulics operating mode and various plant conditions, PREDICT how key system/plant parameters will respond to the following Control Rod Drive Hydraulics component or controller failures:
a. CRD pump trip
b. CRD FCV valve failure open/close
c. Leaking scram valves
d. Excessive accumulator gas pressures
e. Excessive charging water pressure
f. Overpiston flowpath isolated
g. Leaking directional control valves (insert or withdrawal)
h. Low cooling water flow
i. High cooling water pressure

85 | **ID: QDC.ILT.17082** | **Points: 1.00**

(A reference is provided for this question)

Unit 1 is operating at rated power.

A Backwash of the 'B' Clean-Up Filter Demineralizer has just been completed.
Transfer of the RWCU Phase Separator Tank to Radwaste is in progress.
A catastrophic failure of Sludge Mix pump suction line causes a spill into the Reactor Building.
All attempts to isolate the leak have been unsuccessful.

Reactor Building Area Radiation conditions are as follows:

Reactor Building Area Radiation Monitor	Beginning of Shift	Current Conditions
ARM 8 RB RWCU INST AREA	2 mr/hr	3400 mr/hr - In Alarm
ARM 9 RB RWCU PUMP AREA	3 mr/hr	4500 mr/hr - In Alarm
ARM 29 RW PMP RM ACCESS	20 mr/hr	3100 mr/hr - In Alarm
ALL Other Reactor Building ARMs	2 to 5 mr/hr	3 to 7 mr/hr - NOT In Alarm

Which one of the following is the Unit Supervisor required to direct?

A. Commence a Normal Reactor Shutdown to Cold Shutdown IAW QCGP 2-1, NORMAL UNIT SHUTDOWN

B. Evacuate the affected areas IAW QCOA 1800-01, AREA HIGH RADIATION, ONLY

C. Initiate a Manual Scram and enter QGA 100, RPV CONTROL. Emergency Depressurization is NOT required.

D. Initiate a Manual Scram, enter QGA 100, RPV CONTROL, AND perform an Emergency Depressurization IAW QGA 500-1, RPV BLOWDOWN.

Answer: A

Answer Explanation:

The RWCU Phase Separator Tank is not a primary system. With 2 Reactor Building Area radiation levels above Max Safe, QGA 300, Secondary Containment Control, directs that the reactor must be shut down per QCGP 2-1.

Distractor 1 is incorrect: The Reactor cannot stay operating and must be shutdown IAW QGA 300.
Distractor 2 is incorrect: There is no primary system discharging into the Reactor Building.
Distractor 3 is incorrect: There is no primary system discharging into the Reactor Building.

Reference: QGA 300 rev 12

Reference provided during examination: QGA 300 rev 12, with entry conditions removed

Cognitive level: High

Level (RO/SRO): SRO
Tier: 1 **Group:** 2

K/A: 295033.EA2.03 Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS : (CFR 41.10)
Cause of high area radiation (RO=3.7 / SRO=4.2)

SRO Justification: 10 CFR 55.43(b)(4)
Analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures.

Question Source: Hope Creek ILT Exam Bank
Question History: Hope Creek 2009 NRC ILT Exam

Comments:

Associated objective(s):

295033.EA2.03 Cause of high area radiation (RO=3.7 / SRO=4.2)

S-0001-K30 (Freq: LIC=B)
Given QGA 300, 'Secondary Containment Control', and various conditions, EVALUATE the conditions and DESCRIBE how to proceed through the flowchart including transitions to other QGA procedures

86	ID: QDC.ILT.15641	Points: 1.00

A startup is in progress on Unit 1 with the following plant conditions:

- Reactor Mode switch in STARTUP
- IRMs are on Range 8
- Reactor pressure is 135 psig

Which of the following annunciators, if received with the current plant conditions, must be addressed FIRST if the Unit Supervisor places a priority on ensuring the OPERABILITY of Technical Specification required equipment?

(Note: Consider each annunciator separately.)

A. 901-5 A-7, RBM HIGH OR INOP

B. 901-3 A-9, HPCI TURBINE TRIPPED

C. 901-5 D-8, CONTROL ROOM VENT ISOLATED

D. 901-3 C-3, CORE SPRAY PUMP AREA CLR FAN TRIP

Answer: D

Answer Explanation:

901-3 C-3 annunciator indicates that a breaker for the room cooler in the 'A' or 'B' Core Spray pump room is tripped. If the cooler cannot be restored, the Core Spray and/or RCIC subsystem will be required to be declared inoperable.

Distractor 1 is incorrect: Plausible if candidate assumes that the RBM is required per Technical Specifications during a startup. Incorrect because reactor power is < 30% RTP.
Distractor 2 is incorrect: Plausible if candidate assumes that HPCI is required per Technical Specifications during a startup. Incorrect because reactor pressure is less than 150 psig. HPCI is required to be operable per Tech Specs in Modes 1, 2, and 3 with Reactor pressure >150 psig.
Distractor 3 is incorrect: Plausible if candidate assumes that the 'B' train of control room ventilation is impacted by an isolation. A control room isolation will prevent the 'A' (non-safety related) train from running in the normal mode, but will not prevent the 'B' train from satisfying its safety function. Also plausible because reactor building ventilation isolation results in the SRO declaring a Tech Spec required equipment inoperable (rad monitors).

Reference: QCAN 901(2)-3 C-3 rev 10
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): SRO
Tier: 2 **Group:** 1

K/A: 209001 Low Pressure Core Spray System

2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm. (RO=4.1 / SRO=4.3)

Question Source: Quad Cities ILT Exam Bank (QDC.ILT.16413)
Question History: Quad Cities 2011 ILT NRC Exam

SRO Justification: Unique to the SRO position. Candidate must be able to determine operability of Tech Spec required equipment (see facility objective).

Comments: None

Associated objective(s):

S-OPDT-K06 (Freq: LIC=A) Given a degraded or nonconforming condition that may impact the operability of a specific SSC described in Tech Specs, using P&ID/C&IDs, E-prints and Tech Specs, if necessary, PERFORM an immediate Operability Determination and DETERMINE if the SSC meets Tech Spec operability requirements in accordance with the Operability Determination procedures, OP-AA-108-115 and OP-AA-108-115-1002.

209001.2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm. (RO=4.1 / SRO=4.3)

87 **ID: QDC.ILT.17103** **Points: 1.00**

A reactor startup is in progress on Unit 2.

- The Reactor MODE switch is in 'START/HOT STBY'
- All RPS Scram Solenoid Lights are LIT

The following is the present status of the APRM versus LPRM inputs, and indicated power.

APRM:	CH 1	CH 2	CH 3	CH 4	CH 5	CH 6
LPRMs:						
D Level Inputs:	4	5	3	4	5	6
C Level Inputs:	4	3	4	3	4	4
B Level Inputs:	3	4	3	4	5	4
A Level Inputs:	3	3	2	3	1	2
Indicated power:	11%	10%	11%	11%	11%	10%

Based on the above conditions:

What actions are required to maintain compliance with LCO 3.3.1.1, Reactor Protection System Instrumentation?

Declare _(1)_ APRM channel(s) INOPERABLE and insert a half scram on RPS _(2)_ .

A. (1) ONE
(2) 'A'

B. (1) ONE
(2) 'B'

C. (1) TWO
(2) 'A'

D. (1) TWO
(2) 'B'

Answer: C

Answer Explanation:

APRM Channel 3 has only 12 LPRM inputs (<50% / <13) and should have caused an automatic half scram on RPS A. Since all RPS Scram Solenoid Lights are still lit, the automatic scram signal did not actuate. The subsequent actions contained in QCOA 0500-01, Partial Scram Actuation, directs the insertion of a manual scram signal for the affected channel.
APRM channel 5 has only 1 LPRM input on Level 'A', and is therefore administratively inoperable (<2 per level) per QCOP 0700-04, APRM System Operation. This will not cause a half scram to be generated for RPS channel B. However, per the Tech Spec Bases, the APRM must still be declared inoperable, bringing the total number of APRMs required to be declared inoperable up to two.

Distractor 1 is incorrect: Plausible if the candidate does not realize that there are less than the required 'A' level LPRM inputs for APRM 5, or that there is a procedural requirement to declare the the APRM inoperable.
Distractor 2 is incorrect: Plausible if the candidate assumes that APRM 5 should have caused a half scram, and does not realize that APRM 3 should have.
Distractor 3 is incorrect: Plausible if the candidate assumes that APRM 5 should have caused a half scram.

Reference: QCOA 0500-01 rev 8, Tech Spec Bases B 3.3.1.1 rev 0, QCOP 0700-04 rev 16
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): SRO
Tier: 2 **Group:** 1

K/A: 215005.A2.04 Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
SCRAM trip signals (RO=3.8 / SRO=3.9)

Question Source: Modified from Perry 2010 ILT NRC Exam
Question History: N/A

SRO Justification: 10 CFR 55.43(b)(5) This question requires the candidate to use Tech Spec Bases knowledge to assess plant conditions and select appropriate sections of procedures in order to comply with Tech Specs.

Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Comments:

Associated objective(s):

215005.A2.04 SCRAM trip signals (RO=3.8 / SRO=3.9)

S-0703-K33 (Freq: LIC=B)
DISCUSS the bases for LPRM/APRM System Tech Spec LCO, related safety limits and LSSS's

88	ID: QDC.ILT.17109	Points: 1.00

(A reference is provided for this question)

Unit 1 was operating at full power when the 'B' RFP Feed Flow Transmitter failed high. The DFWLCS responded and RPV water level lowered to the scram setpoint.

The Reactor scrammed and RPV water level continued to drop far enough to cause an ECCS Injection signal.

HPCI and RCIC automatically initiated and restored RPV water level.

Which ONE of the following conditions identifies the 10 CFR 50.72 reporting requirements for the above event?

A. Both the Reactor Scram and RPV injection by ECCS were caused by valid signals and are reportable

B. Both the Reactor Scram and RPV injection by ECCS were caused by an invalid signal and are NOT reportable

C. Only the RPV injection by ECCS was caused by a valid signal and is reportable

D. Only the Reactor Scram was caused by a valid signal and is reportable

Answer: A

Answer Explanation:

When the Feed Flow transmitter fails high, the DFWLCS senses a rise in feed flow, causing it to start closing the FRVs. Actual RPV water level will fall, but because the feedwater signal is failed high, DFWLC will initiate Runout Flow Control, preventing the FRVs from opening any further.
The subsequent lowering of actual RPV water level causes a valid RPS signal to scram the reactor on low RPV level (0 inches) and then an ECCS injection signal (-59 inches). Therefore, both the reactor scram and ECCS injection signals are reportable events.

Distractors 1,2 and 3 are incorrect: Plausible because the signal that caused RPV water level to fall was not valid, however, all subsequent effects of the lowering RPV water level are valid.

Reference: LS-AA-1110 rev 16, QCOA 0600-09 rev 11
Reference provided during examination: LS-AA-1110 rev 16

Cognitive level: High

Level (RO/SRO): SRO
Tier: 2 **Group:** 1

K/A: 259002 Reactor Water Level Control System
259002.2.4.30 Knowledge of which events related to system operations/status should be reported to outside agencies (RO=2.2 / SRO=3.6)

Question Source: New

Question History: Modeled after a question on the 2009 Columbia ILT NRC Exam

SRO Justification: 10 CFR 55.43(b)(5)
Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed

Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Comments:

Associated objective(s):

259002.2.4.30 Knowledge of which events related to system operations/status should be reported to outside agencies (RO=2.2 / SRO=3.6)

S-REPT-K02 (Freq: LIC=B NF=B)
Given a reportable event and the appropriate procedure(s), identify the responsibilities of each job position.

89	ID: QDC.ILT.17085	Points: 1.00

Unit 1 was at rated power when operators manually scrammed due to a TOTAL LOSS OF INSTRUMENT AIR.

- All control rods fully inserted
- Aux loads successfully transferred to the RAT
- RPV water level is being maintained by Condensate/Feedwater

A LOCA then developed inside the Drywell along with various containment failures, and the crew subsequently entered QGA 500-1, RPV Blowdown.

The following conditions now exist:

- Only 2 SRVs could be opened and are now open
- RPV to Torus DP is at 300 psig and lowering slowly

Which one of the following procedures is the Unit Supervisor required to direct using in order to depressurize the RPV?

A. QCOP 0250-05, Reactor Pressure Control Using Main Steam Line Drains

B. QCOP 1200-07, RWCU System Coolant Rejection

C. QCOP 1000-05, Shutdown Cooling Operation

D. QCOP 0250-01, Pressurizing The Main Steam Lines

Answer: A

Answer Explanation:

The total loss of Instrument Air will result in eventual closure of the MSIVs and loss of Feedwater. The Main Steam Line Drain Valves are motor operated valves and will be the only available method for depressuring the vessel.

Distractor 1 is incorrect: Plausible because RWCU can normally be used to control reactor pressure, but the Reject Flow Control Valve fails closed on a loss of instrument air.
Distractor 2 is incorrect: Plausible because SDC will be used eventually, however, RPV to Torus DP is given at 300 psig and lowering slowly. Shutdown Cooling cannot be placed in service until RPV pressure is less than 118 psig, which would make Shutdown Cooling unavailable for Emergency Depressurization
Distractor 3 is incorrect: Plausible because this is the normal method for controlling reactor pressure, but the loss of instrument air will close the MSIVs, isolating the reactor from the BPVs.

Reference: QCOP 0220-02, rev 0, QGA 500-1 rev 13
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): SRO
Tier: 2 **Group:** 1

K/A: 300000 Instrument Air System
2.4.06 Knowledge of EOP mitigation strategies. (RO=3.7 / SRO=4.7)

Question Source: New
Question History: N/A

SRO Justification: 10 CFR 55.43(b)(5)
Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed

Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Comments:

Associated objective(s):

300000.2.4.06 Knowledge of EOP mitigation strategies. (RO=3.7 / SRO=4.7)

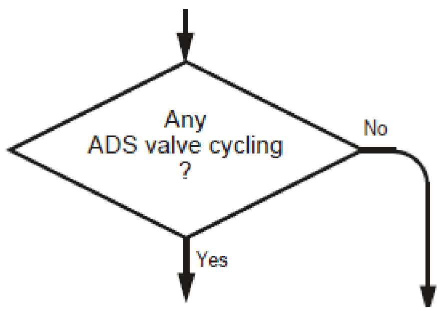
S-0001-K62 (Freq: LIC=B)
Given QGA 101, RPV Control, and various conditions, EVALUATE the conditions and DESCRIBE how to proceed through the flowchart including transitions within QGA 101, to other QGA procedures, to station operating procedures, or to SAMGs.

90 **ID: QDC.ILT.17143** **Points: 1.00**

Unit 1 was at 75% power when there was a complete LOSS of off-site power.

- All rods are fully inserted in the core.
- Reactor Mode switch is in SHUTDOWN.
- The HIGHEST reactor pressure reached during the transient was 1135 psig.
- Reactor pressure is currently 1125 psig and rising at 15 psig per minute.
- HPCI is Out-Of-Service
- QGA 100, RPV Control has been entered but NO further operator action has been taken.

Complete the following statements regarding the execution of the QGA 100 pressure leg:



The correct ANSWER to the decision diamond question "Any ADS Valve Cycling?" is __(1)__.

Given the current plant conditions, the Unit Supervisor must direct the Operator to maintain reactor pressure 800 - 1000 psig using __(2)__.

A. (1) NO
(2) main turbine bypass valves per QCGP 2-3, Reactor Scram

B. (1) NO
(2) ADS valves per QCOP 0203-01, Reactor Pressure Control Using Manual Relief Valve Actuation

C. (1) YES
(2) main turbine bypass valves per QCGP 2-3, Reactor Scram

D. (1) YES
(2) ADS valves per QCOP 0203-01, Reactor Pressure Control Using Manual Relief Valve Actuation

Answer: D

Answer Explanation:

Reactor pressure of 1125 psig is above the lift setpoint of the low-set relief valves (1115 psig). Therefore, at least two ADS valves are expected to by cycling.

With a loss of off-site power, main turbine bypass valves are NOT available (MSIVs are shut and power is lost to the EHC pumps). Therefore, alternate pressure control systems must be used. In this case, the ADS valves will serve as pressure control.

Distractor 1 is incorrect: Combination of distractor 2 and 3.
Distractor 2 is incorrect: Plausible if candidate assumes the high reactor pressure condition did not actuate the low-set relief valves. The other 3 relief valve setpoints are 1135 psig.
Distractor 3 is incorrect: Plausible if candidate does not recognize that the preferred pressure control system (main turbine bypass valves) is not available.

Reference: QGA 100 Rev 9, LIC-0203 Rev 18, QOM 1-6700-T03 Rev 4, QOM 1-6700-T01 Rev 7
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): SRO
Tier: 2 **Group:** 1

K/A: 239002.A2.06 Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
Reactor high pressure (RO=4.1 / SRO=4.3)

Question Source: Modified from 2011 Quad Cities ILT NRC Exam
Question History: 2011 Quad Cities ILT NRC Exam

SRO Justification: Unique to the SRO position
Candidate must assess plant conditions and select appropriate action within the EOPs (see facility objective).

Comments:

Associated objective(s):

239002.A2.06 Reactor high pressure (RO=4.1 / SRO=4.3)

S-0001-K18 (Freq: LIC=B)
Given QGA 100, RPV Control, and various conditions, EVALUATE the conditions and DESCRIBE how to proceed through the flowchart including transitions within QGA 100, to other QGA procedures, station operating procedures, or SAMGs.

91 ID: QDC.ILT.17114 Points: 1.00

Unit 2 is at 30% power. The output of APRM flow converter #1 to the associated Rod Block Monitor (RBM) has failed high.

If a NON-peripheral control rod is selected, which of the following describes the correct action required per Technical Specifications, and the reason for that action?

(Consider each item SEPARATELY)

Declare RBM _(1)_ INOPERABLE to prevent exceeding _(2)_ that may result from a single control rod withdrawal error event.

A. (1) 7
(2) MAPRAT

B. (1) 7
(2) MCPR

C. (1) 8
(2) MAPRAT

D. (1) 8
(2) MCPR

Answer: B

Answer Explanation:

From the Tech Spec bases for Tech Spec 3.3.2.1: The RBM is designed to prevent violation of the MCPR APPLICABILITY SL that may result from a single control rod withdrawal error (RWE) event.

and

The RBM is assumed to mitigate the consequences of an RWE event when operating ☐ 30% RTP and a non-peripheral control rod is selected. Below this power level, or if a peripheral control rod is selected, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE.
The candidate must apply the Tech Spec Bases knowledge of the fuel failure mechanism that can occur during a design basis Rod Withdrawal Error.

Distractor 1 is incorrect: Plausible if candidate assumes that the RBM protects against violating MAPRAT, which is a protection limit that ensures that the peak cladding temperature following a design basis LOCA will not exceed 2200 degrees F.
Distractor 2 is incorrect: Combination of distractors 1 and 3.
Distractor 3 is incorrect: Plausible if the candidate assumes that RBM 8 is supplied with the output from the #1 flow converter.

Reference: Tech Spec Bases B 3.3.2.1 (Rev 0 page 3, Rev 43 page 4) LIC 0705 rev 9
Reference provided during examination: None

Cognitive level: Memory

Level (RO/SRO): SRO
Tier: 2 **Group:** 2

K/A:
215002 Rod Block Monitor System
215002.2.2.40 Ability to apply technical specifications for a system. (RO=3.4/ SRO=4.7)

SRO Justification: 10 CFR 55.43(b)(2) Application of the Bases in technical specifications
Can question be answered *solely* by knowing ≤ 1 hour TS/TRM Action?
NO
Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?"
NO
Can question be answered *solely* by knowing the TS Safety Limits?
NO

Question Source: New
Question History: N/A

Comments:

Associated objective(s):

S-0705-K33 (Freq: LIC=B)
DISCUSS the bases for Rod Block Monitor System Tech Spec LCOs.

215002.2.2.40 Ability to apply technical specifications for a system. (RO=2.9 / SRO=4.0)

92 **ID: QDC.ILT.17087** **Points: 1.00**

A steam line break in primary containment has caused a reactor scram. Twelve control rods are NOT full in. Reactor water level was 55 inches when erratic level indication was observed on all available level indications during depressurization.

Which one of the following predicts the response of the available level indication as the reactor depressurizes and also identifies the required procedures when reactor water level CAN NOT be determined?

Indicated RPV level will eventually fail __(1)__.

The Unit Supervisor is required to __(2)__.

A. (1) downscale
(2) exit QGA 101 AND enter QGA 500-4, RPV FLOODING

B. (1) downscale
(2) execute only the power leg of QGA 101 AND enter QGA 500-4, RPV FLOODING

C. (1) upscale
(2) exit QGA 101 AND enter QGA 500-4, RPV FLOODING

D. (1) upscale
(2) execute only the power leg of QGA 101 AND enter QGA 500-4, RPV FLOODING

Answer: D

Answer Explanation:

The given indications describe RPV level instruments in the saturation condition, reducing the amount of water in the reference leg, which results in a lower pressure on that side of the transmitter, therefore indicated level is higher than actual.
The override in QGA 100 states to exit the procedure and enter QGA 500-4 when RPV water level is unknown.
The override in QGA 101 states to exit only the Pressure and Level legs of the procedure and enter QGA 500-4 when RPV water level is unknown.

Distractor 1 is incorrect: The instruments will fail upscale. QGA 101 is not exited completely.
Distractor 2 is incorrect: The instruments will fail upscale.
Distractor 3 is incorrect: QGA 101 is not exited completely.

Reference: QGA 100 rev 9, QGA 101 rev 13
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): SRO
Tier: 2 **Group:** 2

K/A: 216000.A2.07 Ability to (a) predict the impacts of the following on the NUCLEAR BOILER INSTRUMENTATION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
Reference leg flashing (RO=3.4 / SRO=3.5)

Question Source: Brunswick ILT Exam Bank
Question History: Brunswick 2008 ILT NRC Exam

SRO Justification: 10 CFR 55.43(b)(5)
Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed

Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Comments:

Associated objective(s):

216000.A2.07 Reference leg flashing (RO=3.4 / SRO=3.5)

SR-0263-K22 (Freq: LIC=B)
Given a RPV Instrumentation System operating mode and various plant conditions, PREDICT how the RPV Instrumentation System will respond to the following failures/conditions:
a. RPV pressure not at instrument's calibration pressure
b. Changes in steam flow
c. Changes in recirc flow / RHR injection flow
d. Normal RPV cooldown
e. Rapid RPV depressurization below 450 psig
f. Elevated drywell/reactor building temperatures
g. Reference / Variable leg leaks
h. Sudden increase in RVLIS flow/pressure
i. RPV head seal leak
j. RPV temperature stratification

93 **ID: QDC.ILT.17089** **Points: 1.00**

(A reference is provided for this question)

Given:

- Unit 1 is in Day 14 of a refueling outage
- RPV water level is 23.5 feet above the top of the RPV flange
- A loss of all Instrument Air has occurred
- Annunciator 901-3 D-7, RHR HX COOLING WATER/FUEL POOL HX HIGH TEMP, is in alarm
- Point 6 (Fuel Pool Cooling Pump Suction) on TR 1-1040-5 at the 901-3 panel reads 125.6ºF and rising

Which one of the following actions is the Unit Supervisor required to direct in order to control Fuel Pool Temperature?

Manually throttle the __(1)__ to maintain cooling flow to the Fuel Storage Pool IAW Fuel Pool Cooling Abnormal Procedure __(2)__.

A. (1) 1-1901-58, FUEL POOL TO CCST FCV, OPEN
(2) QCOA 1900-02, Fuel Storage Pool High Temperature

B. (1) 1-1901-58, FUEL POOL TO CCST FCV, OPEN
(2) QCOA 1900-01, Loss Of Water Level In The Fuel Storage Pool Or Reactor Cavity

C. (1) 1-1901-40, FUEL POOL F/D BYP VLV, OPEN
(2) QCOA 1900-02, Fuel Storage Pool High Temperature

D. (1) 1-1901-40, FUEL POOL F/D BYP VLV, OPEN
(2) QCOA 1900-01, Loss Of Water Level In The Fuel Storage Pool Or Reactor Cavity

Answer: C

Answer Explanation:

As shown on print M-45, a loss of Instrument Air causes the Air operated Fuel Pool Filter Demin Flow Control Valves to fail closed, stopping all flow through the demins, leaving the only flowpath (shown on print M-38) to the fuel pool through the manually operated and normally cracked open 1-1901-40, FUEL POOL F/D BYP VLV. This drastically reduces the amount of flow from the FPC pumps to the fuel pool, resulting in a rapid rise in temperature. At 125ºF FPC pump suction temperature, the US will enter QCOA 1900-02, Fuel Storage Pool High Temperature, and direct the 1-1901-40 valve to be throttled open to maintain cooling flow.

Distractor 1 is incorrect: Plausible because second part of distractor is correct. There is still flow from the pumps and an alternate source is not needed.
Distractor 2 is incorrect: Plausible because this would be a correct answer if a loss of the FPC pumps had occurred. The isolation of the FPC demins will not cause the FPC pumps to trip.
Distractor 3 is incorrect: Plausible because first part of distractor is correct. The fuel pool cooling pumps did not trip.

Reference: QCOA 1900-02 rev 7, QCAN 901(2)-3 D-7 rev 4, M-38
Reference provided during examination: M-38 and M-45

Cognitive level: High

Level (RO/SRO): SRO
Tier: 2 **Group:** 2

K/A: 233000.A2.05 Ability to (a) predict the impacts of the following on the FUEL POOL COOLING AND CLEAN-UP ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR 41.5 / 45.6)
Valve closures (RO=2.5 / SRO=2.5)

SRO Justification: 10 CFR 55.43(b)(5)
Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed

Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? NO

Can the question be answered solely by knowing immediate operator actions? NO

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? NO

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? NO

Question Source: New
Question History: N/A

Comments:

Associated objective(s):

233000.A2.05 Valve closures (RO=2.5 / SRO=2.5)

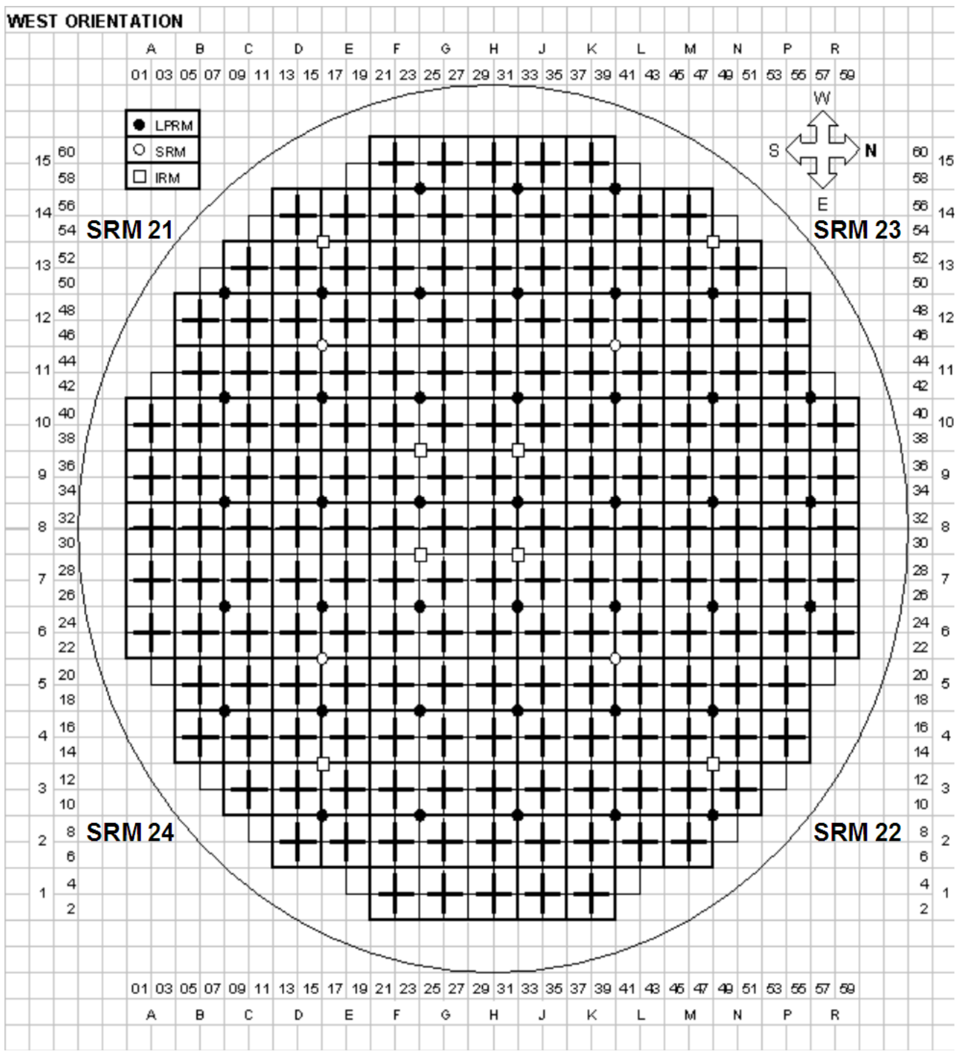
SRN-1900-K23 (Freq: LIC=B NF=B)
Given a Fuel Pool Cooling System operating mode and various plant conditions, PREDICT how the Fuel Pool Cooling System will be impacted by the following support system failures:
a. Loss of 480 vac
b. Loss of service air
c. Loss of condensate transfer
d. Loss of clean demin
e. Loss of RHR/RHRSW
f. Loss of RBCCW
g. Condensate phase separators
h. Reactor Building ventilation
i. Loss of instrument air

94 **ID: QDC.ILT.15665** **Points: 1.00**

Given:

- Unit 1 is in a refueling outage with fuel offloading scheduled to begin this shift.
- SRMs 22, 23 and 24 are INOPERABLE.

You are facing the 901-5 panel full-core display.



Which of the following is true in regards to the Technical Specification requirements for fuel moves?

A. Fuel can be offloaded from ALL quadrants.

B. Fuel can be offloaded from all quadrants except the LOWER RIGHT.

C. Fuel can be offloaded from the UPPER LEFT quadrant ONLY.

D. Fuel can NOT be offloaded from ANY quadrant.

Answer: D

Answer Explanation:

QCFHP 0100-01, Step E.5.d, states "only one SRM channel is required to be OPERABLE during spiral offload or reload when the fueled region includes only that SRM detector".
However, with fuel in all quadrants, two SRMs are required to be operable, one in the quadrant where fuel is to be moved, and one in the adjacent quadrant.

Distractor 1 is incorrect: Plausible if the candidate assumes that only one SRM is required to be operable when performing a core offload.
Distractor 2 is incorrect: Plausible if candidate assumes that an SRM in an adjacent quadrant meets the requirements.
Distractor 3 is incorrect: Plausible if candidate assumes that only one SRM channel is required to be operable if the offload takes place in the fueled region that the SRM is in.

Reference: QCFHP 0100-01 Rev 32, Tech Specs 3.3.1.2 amendment 199/195
Reference provided during examination: N/A

Cognitive level: High

Level (RO/SRO): SRO
Tier: 3 **Group:** N/A

K/A: 2.1 Conduct Of Operations
2.1.41 Knowledge of the refueling process. (RO=2.8 / SRO=3.7)

Question Source: Modified from Quad Cities 2011 ILT NRC Exam
Question History: N/A

SRO Justification: 10 CFR 55.43(b)(6)
Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

Knowledge of TS bases for reactivity controls.

Comments: None.

Associated objective(s):

SRL-805-K19 (Freq: LIC=I NF=I)
Given Refueling related equipment operability or key parameter indications and various plant conditions, DETERMINE, from memory, if the Conduct of Refueling Tech Spec LCOs have been met.

2.1.41 Knowledge of the refueling process. (RO=2.8 / SRO=3.7)

95	ID: QDC.ILT.15668	Points: 1.00

(A reference is provided for this question)

Unit 1 is at 100% power.

- The 1A RHRSW Pump is declared inoperable at 0300 on June 1
- The 1B RHRSW Pump is declared inoperable at 1000 on June 2

Which ONE of the following is required to maintain compliance with Technical Specifications?

A. Enter LCO 3.0.3 and begin a Shutdown Time Clock by 1100 on June 2

B. Enter LCO 3.7.1 Condition B and begin a Shutdown Time Clock by 1000 on June 9

C. Enter LCO 3.7.1 Condition C and begin a Shutdown Time Clock by 1000 on June 9

D. Enter LCO 3.7.1 Condition C and begin a Shutdown Time Clock by 1000 on June 10

Answer: C

Answer Explanation:

Examinee must recognize that 1A and 1B RHRSW pumps are in the same subsystem (pump numbering does NOT follow conventional numbering scheme) and therefore NO extensions apply.
Per the Tech Spec bases, both pumps in the same subsystem must be operable for the subsystem to be operable. When the first pump goes inoperable, start a 30-day clock per LCO 3.7.1 Required Action A.1.
The second pump going inoperable does NOT affect the 30 day time clock.
The clock will expire 30 days after the first pump goes inoperable (0300 on July1), after which LCO 3.7.1 condition D would apply, starting the shutdown time clock.
However, when the second pump goes inoperable, LCO 3.7.1 condition C is entered because the subsystem will be inoperable due to a reason other than ONE pump being inoperable.
Action C.1 requires the subsystem restored to operable status within 7 days, after which condition D will apply, starting the shutdown time clock.

Note: LCO 3.4.7, RHR Shutdown Cooling System - Hot Shutdown, is NOT applicable in Mode 1.

Distractor 1 is incorrect: Plausible if candidate assumes LCO 3.0.3 applies. LCO 3.0.3 is NOT applicable because LCO 3.7.1 Condition C allows for the subsystem to be declared inoperable for reasons other than ONE pump being inoperable. This would be the correct answer if LCO 3.0.3 applied.

Distractor 2 is incorrect: Plausible if the candidate does not recognize the pumps are in the same subsystem. This would be the correct answer if the pumps were not in the same subsystem.

Distractor 3 is incorrect: Plausible if the candidate incorrectly applies the extension time of 24 hours to the correct LCO Condition. This would be the correct answer if the extension were allowed.

Reference: Technical Specification 3.7.1, Tech Spec Bases B.3.7.1
Reference provided during examination: Technical Specification 3.7.1 pages 3.7.1-1 and 3.7.1-2 with LCO statement and Applicability statement removed.

Cognitive level: High

Level (RO/SRO): SRO
Tier: 3 **Group:** N/A

K/A: 2.2 Equipment Control
2.2.23 Ability to track Technical Specification limiting conditions for operations (RO=4.0 / SRO=4.7)

Question Source: New
Question History: N/A

SRO Justification: 10 CFR 55.43(b)(2)
This question requires the candidate to have knowledge of TS bases that is required to analyze TS required actions and terminology

Can question be answered *solely* by knowing ≤ 1 hour TS/TRM Action?
NO
Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?"
NO
Can question be answered *solely* by knowing the TS Safety Limits?
NO

Comments:

Associated objective(s):

2.2.23 Ability to track Technical Specification limiting conditions for operations (RO=4.0 / SRO=4.7)

S-1000-K33 (Freq: LIC=B)
Discuss the bases for RHR/RHRSW System LCO's.

96	ID: QDC.ILT.15666	Points: 1.00

Complete the following statement regarding the BASES for the ACTIONS REQUIRED when a Tech Spec Safety Limit is violated.

Exceeding a Safety Limit may cause __(1)__ and create the potential for excessive release rates. Therefore, all insertable control rods must be inserted __(2)__.

A. (1) fuel damage
(2) immediately

B. (1) fuel damage
(2) within two hours

C. (1) containment failure
(2) immediately

D. (1) containment failure
(2) within two hours

Answer: B

Answer Explanation:

TS Bases SL 2.2 Violation:

Exceeding a Safety Limit may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 50.6 limits. Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

Distractor 1: Incorrect but plausible. The second part of the answer is incorrect.
Distractor 2: Combination of distractors 1 and 2.
Distractor 3: Incorrect but plausible. The containment structure is the last barrier to fission product release.

Reference: Reactor Core Safety Limit Bases Section B 2.1.1
Reference provided during examination: N/A

Cognitive level: Memory

Level (RO/SRO): SRO
Tier: 3 **Group:** N/A

Question Source: Quad Cities ILT Exam Bank (Question ID 14273)
Question History: N/A

SRO Justification: 10 CFR 55.43(b)(2)

Can question be answered *solely* by knowing ≤ 1 hour TS/TRM Action?
NO
Can question be answered *solely* by knowing the LCO/TRM information listed “above-the-line?”

NO
Can question be answered *solely* by knowing the TS Safety Limits?
NO

Does the question involve one or more of the following for TS, TRM, or ODCM?

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)

Knowledge of TS bases that is required to analyze TS required actions and terminology

Comments:

Associated objective(s):

2.2.25 Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (RO=3.2 / SRO=4.2)

S-0201-K33 (Freq: LIC=I)
DISCUSS the bases for Reactor Vessel and Internals Tech Spec LCO, related safety limits and LSSS's

97	ID: QDC.ILT.15658	Points: 1.00

Given:

- A severe natural disaster has occurred resulting in a complete loss of cooling to both unit's fuel pools.
- An explosion has caused many fires to develop and part of the refuel floor roof to collapse.
- One of the fire-brigade members has been injured and is unconscious.
- Because of the nature and progression of the fire, his life is in imminent danger.
- Radiation levels in the area are 50 R/hr.

Acting as the Station Emergency Director, you have been requested to authorize Emergency Exposure Limits to allow entry into an area to **save the life** of the fire-brigade member.

A volunteer, who is **FULLY AWARE** of the risks involved, has agreed to attempt the rescue. The volunteer is an adult male, Exelon employee (Occupational Worker) who does NOT have high lifetime exposure, AND has NOT had any Planned Special Exposures OR administrative increases in his exposure limits.

Complete the following two statements regarding the volunteer's Total Effective Dose (TEDE) limits.

PRIOR TO this event, company policy administratively limits him to an ANNUAL limit of _(1)_ .
DURING this event, the MAXIMUM time you can authorize the volunteer to be in the area is _(2)_ .

A. (1) 2 REM
(2) 30 minutes

B. (1) 2 REM
(2) unlimited

C. (1) 5 REM
(2) 30 minutes

D. (1) 5 REM
(2) unlimited

Answer: B

Answer Explanation:

The Exelon administrative limit is 2000 mrem routine cumulative TEDE/yr for adults. For a lifesaving activity with the worker fully aware of the risks involved, the limit for dose exposure is > 25 REM. Therefore there is no time limit

Distractor 1 is incorrect: 25 REM is the lifesaving limit for individuals NOT fully aware of the risks involved with the radiation exposure.
Distractor 2 is incorrect: Combination of distractor 1 and 3.
Distractor 3 is incorrect: 5 REM is the NRC annual limit.

Reference: RP-AA-203 Rev 3
Reference provided during examination: N/A

Cognitive level: Memory

Level (RO/SRO): SRO
Tier: 3 **Group:** N/A

Question Source: Modified from 2009 ILT NRC EXAM
Question History: N/A

10 CFR Part 55 Content: 43(b)(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures.

Comments: None

Associated objective(s):

SRNLF-00-K08 (Freq: LIC=I NF=I)
Given symptoms and indications depicting a generic abnormal condition and the administrative procedures, DESCRIBE the operator actions of the applicable administrative procedure.

2.3.04 Knowledge of radiation exposure limits under normal or emergency conditions. (RO=3.2 / SRO=3.7)

98	ID: QDC.ILT.15649	Points: 1.00

A transient has occurred and the SAMGs have been entered. A maintenance team is to be dispatched from the OSC to repair vital equipment. The area to be entered by the team is suspected to have a high iodine concentration.

If Potassium Iodide is issued to personnel on the maintenance team, which of the following individuals MUST authorize its use?

A. OSC Director

B. Station Emergency Director

C. Radiation Protection Manager

D. Occupational Health Services Manager

Answer: B

Answer Explanation:

Per EP-AA-113, section 4.4.2, Authorization. Step 1 states,"The Station Emergency Director must authorize issuance of KI to Exelon emergency workers".

Distractor 1: Plausible because the OSC Director can recommend the issuance of KI, but not approve it.

Distractor 2: Plausible because the Radiation Protection Manager can authorize exposure to less than 5 Rem TEDE and can recommend the issuance of KI, but not approve it.

Distractor 3: Plausible because the Occupational Health Services Department must be notified if KI is to be issued to Exelon Nuclear personnel or contractors.

Reference: EP-AA-113 rev 10
Reference provided during examination: None

Cognitive level: Memory

Level (RO/SRO): SRO
Tier: 3 **Group:** N/A

Question Source: Quad Cities ILT Exam Bank (QDC.ILT.15453)
Question History: Quad Cities 2009 ILT Cert Exam

SRO Justification: Issuance of KI is an SRO ONLY function
10 CFR 55.43(b)(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures.

Comments: None

Associated objective(s):

2.3.13 Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (RO=3.4 / SRO=3.8)

SRNLF-00-K06 (Freq: LIC=I NF=I)
Given an administrative procedure, DESCRIBE the responsibilities of the different job positions required to complete the procedure. (i.e.,Who does what?)

99 **ID: QDC.ILT.17116** **Points: 1.00**

A complete Loss of Offsite Power (LOOP) has occurred on Unit 1. QCGP 2-3, Reactor Scram, and QCOA 6100-03, LOOP, are being performed.

Torus water temperature is 96ºF and rising.

Which ONE of the following describes the required actions to be directed by the Unit 1 Unit Supervisor?

A. Concurrently enter QGA 200, Primary Containment Control, and start all available Torus cooling.

B. Exit QCOA 6100-03, enter QGA 200, Primary Containment Control and start all available Torus Cooling.

C. Continue QCOA 6100-03 and QCGP 2-3, monitor Torus water temperature, and DO NOT enter QGA 200, Primary Containment Control.

D. Exit QCGP 2-3, enter QGA 200, Primary Containment Control, monitor Torus water temperature, and DO NOT start all available Torus cooling.

Answer: A

Answer Explanation:

When a QGA entry condition is reached, the QGA is entered. QCOA execution supplements QGA execution because the QGAs are symptom based, not event based. The actions of QGA 200 must be directed, and will take precedence over concurrent QCOAs if necessary, but there is no explicit direction to exit QCOAs when entering QGAs.

Distractor 1 is incorrect: No explicit direction exists to exit the QCOA and such action would be detrimental to restoring electrical power sources.
Distractor 2 is incorrect: While continuing in QCOAs is correct, once a QGA entry condition is reached, entering the QGA is required immediately.
Distractor 3 is incorrect: No explicit direction exists to exit the QCOA. Actions of the QGAs must be directed once the QGAs are entered.

Reference: QCAP 0200-10, EMERGENCY OPERATING PROCEDURE (QGA) EXECUTION STANDARDS (Rev 42)
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): SRO
Tier: 3

K/A: 2.4 EMERGENCY PROCEDURES / PLAN
2.4.08 Knowledge of how abnormal operating procedures are used in conjunction with EOPs. (RO=3.8 / SRO=4.5)
10 CFR 55.43 part 5

SRO Justification: 10 CFR 55.43(b)(5) Assessment of the situation provided in the stem and selection of an appropriate procedure (QGA 200)

Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed

Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Question Source: New
Question History: N/A

Comments: Modeled after a question on the LaSalle 2008 ILT NRC Exam

Associated objective(s):

S-0001-K24 (Freq: LIC=B)
Given QGA 200, 'Primary Containment Control' and QGA 200-5, 'Hydrogen Control', and various conditions, EVALUATE the conditions and DESCRIBE how to proceed through the flowcharts including transitions within QGA 200 or 200-5, to other QGA procedures or to normal operating procedures.

2.4.08 Knowledge of how abnormal operating procedures are used in conjunction with EOPs. (RO=3.8 / SRO=4.5)

100	ID: QDC.ILT.17144	Points: 1.00

Unit 1 has a steam leak in the Drywell. Drywell sprays were initiated and secured when required. Containment parameters are presently:

- Drywell temperature 120°F
- Drywell pressure 1.80 psig and rising
- Torus temperature 91°F
- Torus pressure 1.80 psig and rising

The NSO reports the following annunciators are in alarm:

- Annunciator 901-3 C-13, "TORUS VACUUM BRK VALVES OPEN DIV I".
- Annunciator 901-3 G-11, "TORUS VACUUM BRK VALVES OPEN DIV II".

What is the impact, if any, to the primary containment and why?

The primary containment...

A. CAN perform its intended safety function. There is NO impact on primary containment operability.

B. may NOT perform its intended safety function because initial conditions are NOT met for ensuring the maximum drywell pressure during a LOCA will remain below the design value.

C. may NOT perform its intended safety function because initial conditions are NOT met for ensuring the negative differential pressure across the drywell wall will remain below the design value.

D. may NOT perform its intended safety function because initial conditions are NOT met for ensuring that an event producing hydrogen and oxygen does NOT result in a combustible mixture inside the primary containment.

Answer: B

Answer Explanation:

Technical Specification bases states that all drywell-to-torus vacuum breakers must be closed to satisfy the pressure-suppression function of the containment.

With annunciators 901-3 C-13(G-11) in alarm, one of the 12 Drywell-Torus vacuum breakers is open and compromising the pressure-suppression function of containment.

Candidate must verify that the alarms for the vacuum breakers are consistent with plant conditions (d/p is zero between the drywell and torus when it should be no less than 0.5 psid).

Distractor 1 is incorrect: Plausible to consider primary containment operable because not all 12 vacuum breakers have to open as intended. Also plausible if candidate assumes that there is a second vacuum breaker in the line (similar to torus-to-reactor building vacuum breakers).
Distractor 2 is incorrect: Plausible because the assumptions for closed vacuum breakers control the amount of negative d/p across the containment walls.

Distractor 3 is incorrect: Plausible if candidate confuses the alarm indication for the torus-to-reactor building vacuum breakers (i.e. D-14, Torus Vacuum Relief LVL 20B Not Closed)

Reference: TS B 3.6.1.8 Rev 40, QCAN 901(2)-3 C-13 rev 11, QCAN 901(2)-3 G-11 rev 9
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): SRO
Tier: 3

K/A: 2.4 EMERGENCY PROCEDURES / PLAN
2.4.46 Ability to verify that the alarms are consistent with the plant conditions. (RO=4.2 / SRO=4.2)

Question Source: Quad Cities 2011 ILT NRC Exam
Question History: Quad Cities 2011 ILT NRC Exam

SRO Justification: 10 CFR 55.43(b)(2) Candidate must have knowledge of Technical Specification bases and primary containment operability requirements.

Can question be answered solely by knowing = 1 hour TS/TRM Action?
NO
Can question be answered solely by knowing the LCO/TRM information listed "above-the-line?"
NO
Can question be answered solely by knowing the TS Safety Limits?
NO

Does the question involve one or more of the following for TS, TRM, or ODCM?

Knowledge of TS bases that is required to analyze TS required actions and terminology

Comments: None

Associated objective(s):

S-1601-K33 (Freq: LIC=I)
DISCUSS the bases for Containment Systems Tech Spec LCO's.

2.4.46 Ability to verify that the alarms are consistent with the plant conditions. (RO=4.2 / SRO=4.2)