Question: 1

Unit 1 Reactor Recirculation Pump 1A tripped.

- 1B Reactor Recirculation Pump speed has been lowered to provide 45% rated core flow.
- The 1-0202-5A, PMP DISCH VLV, has been reopened.

___(1)____ of the Jet Pump Loop A flow indicated on the 901-4 panel is going through the core.

___(2)____ of the flow indicated on the B loop is going through the core.

A. (1) A small fraction
(2) All

B. (1) A small fraction
(2) Most

C. (1) None
(2) All

D. (1) None
(2) Most

Answer: __D__

Explanation:

A) Incorrect. The small fraction of flow indicated in the A loop is reverse flow which is bypassing the core. Therefore, not all flow is passing through the core and no flow indicated in loop A is passing through the core.
B) Incorrect. The small fraction of flow indicated in the A loop is reverse flow which is bypassing the core.
C) Incorrect. None of the flow indicated in the A loop is passing through the core, but since reverse flow through the A loop is bypassing the core, not all indicated flow is passing through the core.
D) Correct. In single-loop operation with the running pump at > approx. 32% speed, the indicated flow through the jet pumps on the idle loop is REVERSE flow. The running recirc pump is providing flow through the core (majority of flow) and some reverse flow through the idle jet pumps which bypasses the core.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295001AK2.07	
	Importance Rating	3.4	

K/A Statement: PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION and the following: core flow indication (CFR 41.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.) QCOA 0202-04, Reactor Recirc Pump Trip – Single Pump, Rev. 46

Proposed references to be provided to applicants during examination: None

Learning Objective: SRN-0202-K20

Question Source: Bank # X
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	__x__
Level of difficulty:	(1-5)	__2__
10 CFR Part 55 Content:	55.41	__7__
	55.43	______

Comments:

Question: 2

The plant was operating at rated conditions when a Station Blackout occurred. A short time later, the following annunciator is received: HPCI GRP 4 PCI VLVS DC DIV ISOL.

Given the above, which one of the following strategies is to be used for RPV pressure control?

A. Rapidly depressurize the RPV with manual Relief Valve actuation to cooldown the reactor.

B. Maintain RPV pressure between 940 and 1060 psig with manual Relief Valve actuation.

C. Maintain RPV pressure between 940 and 1060 psig with turbine bypass valves.

D. Defeat the HPCI Steam Supply Isolation signals, then maintain RPV pressure 940 and 1060 psig with HPCI.

Answer: B

Explanation:

A) Incorrect. Depressurizing removes the ability of HPCI or RCIC to provide level control. This is plausible because a scram has occurred and one of the actions after a scram without the main condenser (per QCGP-2.3 step F.6.c) available is to depressurize the RPV to cooldown the reactor.
B) Correct. The SBO procedure (QCOA-6100-4) step D.6 directs using the QGAs to control reactor pressure. The Pressure control leg of RPV control (QGA-100) directs the use of alternate pressure control methods since the turbine bypass valves are unavailable to control pressure between 940 and 1060 psig.
C) Incorrect. The Pressure control leg of RPV control (QGA-100) directs pressure to be maintained less than 1060 psig using turbine bypass valves, but the condenser and the turbine bypass valves are not available due to the SBO.
D) Incorrect. Although HPCI would be preferred over manual SRV actuation because it can control level, the annunciator indicates that HPCI is now isolated and is not usable without further action for pressure control. (See QCAN 901(2)-3 D-10 and DGA 100).

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295003AK1.06	
Importance Rating	3.8	

K/A Statement: Partial or Complete Loss of AC Power: Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Station blackout: Plant-Specific. (CFR: 41.8 to 41.10)

Technical Reference(s):
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (as available)

Question Source:
Bank # ______
Modified Bank # X (Note changes or attach parent)
New ______

Question History: Last NRC Exam Monticello 2013 Question #2

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis X

Level of Difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	8-10
	55.43	

Comments:

Changes from the Monticello question were to change answer (D) to HPCI and add in the step an annunciator which denotes that HPCI has isolated.

Question 3

Unit 1 is operating at 100% power when Annunciator 901-8 B-9, 125V BATTERY GROUND, ALARMS.

(1) The magnitude of the ground can be determined using ground detection instrumentation located in the ….

(2) ___________ must be located and isolated immediately.

A. (1) Battery Charger room
(2) All Grounds, regardless of magnitude,

B. (1) Battery Charger room
(2) Only grounds of less than or equal to 125,000 Ohms

C. (1) Main Control Room on the 901-8 Panel
(2) All Grounds, regardless of magnitude,

D. (1) Main Control Room on the 901-8 panel
(2) Only grounds of less than or equal to 125,000 Ohms

Answer: B

Explanation:

QOP 6900-06, 125 VDC Ground Detection procedure step F.2 indicates magnitude of the ground is determined using the ground detector recorder which is located in the battery charger room (Turbine Bldg MCC 1A). QCOP 6900-19, Documenting 125/250 VDC Grounds, step F.3 states that Level I (> 125,000 Ohms) grounds need to be documented, whereas Level II and Level III (LTE 125,000 Ohms) need to be located and isolated immediately; additionally Level III grounds must be repaired within 14 days.

A) Incorrect: (1) Recorder is located in the Battery Charger room. (2) Only Level II and Level III grounds need to be isolated immediately. Plausible since all grounds present some level of risk of equipment damage.
B) Correct: (1) Ground magnitude must be read locally in the Battery Charger room. (2) Only Level II and Level III (LTE 125,000 Ohms) need to be located and isolated immediately.
C) Incorrect: (1) Ground magnitude must be read locally in the Battery Charger room. Plausible since other Battery and DC Bus parameters are available in the MCR. (2) Only Level II and Level III (LTE 125,000 Ohms) need to be located and isolated immediately. Plausible since all grounds present some level of risk of equipment damage.
D) Incorrect: (1) Ground magnitude must be read locally in the Battery Charger room. Plausible since other Battery and DC Bus parameters are available in the MCR. (2) Level II and Level III (LTE 125,000 Ohms) need to be located and isolated immediately.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295004AK3.02	
Importance Rating	2.9	

K/A Statement: PARTIAL OR COMPLETE LOSS OF D.C. POWER: Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: ground isolation/fault determination (CFR 41.5)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QCOP 6900-19, Documenting 125/250 VDC Grounds, Rev. 15
2. QOP 6900-06, 125 Volt DC Ground Detection Unit 1 – Single Circuit Isolation, Rev. 58
3. LN-6900, Battery and DC Distribution Lesson Plan; Rev 021

Proposed references to be provided to applicants during examination: None

Learning Objective: SRN-6900-K28

Question Source:

Bank #	
Modified Bank #	(Note changes or attach parent)
New	X

Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
Level of difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	5
	55.43	

Comments:

Question: 4

Main Turbine TRIP pushbuttons (2 at each location) are located in the Control Room and at the __(1)____.

The MINIMUM action to trip the turbine requires ___(2)____ (at the associated location) to be depressed.

A. (1) Main Turbine front standard
(2) BOTH pushbuttons

B. (1) Main Turbine front standard
(2) EITHER pushbutton

C. (1) DEHC cabinet in the Aux Electric Room
(2) BOTH pushbuttons

D. (1) DEHC cabinet in the Aux Electric Room
(2) EITHER pushbutton

Answer: A

Explanation:

A) Correct. (1) The turbine trip pushbuttons are located at the front standard. (2) The turbine trip logic requires both buttons to be depressed to actuate the trip.
B) Incorrect. (1) This part is correct. The turbine trip pushbuttons are located at the front standard. (2) Plausible because the Reactor trip pushbuttons only require either button to be depressed.
C) Incorrect. (1) Plausible because tripping the EHC pumps will also trip the turbine which can be done from the EHC cabinet. (2) This part is correct. The turbine trip logic requires both buttons to be depressed to actuate the trip.
D) Incorrect. (1) Plausible because tripping the EHC pumps will also trip the turbine which can be done from the EHC cabinet. (2) Plausible because the Reactor trip pushbuttons only require either button to be depressed.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295005G2.1.30	
Importance Rating	4.4	

K/A Statement: Main Turbine Generator Trip: Ability to locate and operate components, including local controls. (CFR: 41.7 / 45.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
1. LIC 5600 Main Turbine & Auxiliaries, Rev. 19

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-5600-K16 STATE the physical location and DESCRIBE the operation of the following Main Turbine Control - EHC Hydraulic System controls (local/remote):

a. EHC pump control switches
b. EHC pump test start pushbutton
c. EHC filter pump control switch
d. Main turbine supervisory trip cutout switch
e. EHC heater/fan controls

Question Source:
Bank # X
Modified Bank # _______ (Note changes or attach parent)
New _______

Question History: Last NRC Exam From the licensee's training bank.

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	____
Level of Difficulty:	(1-5)	2
10 CFR Part 55 Content:	55.41	7
	55.43	____

Comments: Made some minor stylistic changes to the question to more closely match the other portions of the exam.

Question: 5

Which of the following conditions will NOT generate an automatic RPS actuation?

A. Turbine trip at 42% reactor power

B. Steam tunnel temperature rises to 210ºF in MODE 1

C. Reactor power spikes to 21% in MODE 2

D. Torus water level lowered to 10.5 feet in MODE 2

Answer: D

Explanation:

A) Incorrect. With the reactor producing steam at greater than 38.5% (as measured by turbine first stage pressure) the RPS scram logic is no longer bypassed, and an automatic reactor scram will be inserted.
B) Incorrect. With steam tunnel ambient temperature greater than 200ºF the Main Steam Isolation valves will receive a closed signal. With the MSIV's less than 90% open, RPS will insert an automatic scram signal.
C) Incorrect. With the MODE switch in MODE 2, the APRM scram signal is enforced at 20% reactor power. This will insert an automatic reactor scram.
D) Correct. With torus water level below 11 feet and in MODE 2, QGA 200 will require the Operating crew to insert a manual reactor scram. There are no automatic reactor scram signals on Torus water level.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295006AA1.01	
Importance Rating	4.2	

K/A Statement: SCRAM: Ability to operate and/or monitor the following as they apply to SCRAM: RPS (CFR 41.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
1. QGA 200, Primary Containment Control, Rev. 11
2. QCGP 2-3, Reactor Scram, Rev. 87

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0500-K07

Question Source:
Bank # X
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam Duane Arnold NRC ILT Exam 2017

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ____

Level of difficulty: (1-5) 2

10 CFR Part 55 Content:
55.41 7
55.43 ____

Comments:

Question: 6

Unit 1 is operating at 100% power.

- A fire breaks out in the Panel 901-6.
- The Unit Supervisor (US) enters QOA 0010-05 Plant Operation with the Control Room Inaccessible.
- The UNSO inserts a manual Reactor SCRAM prior to leaving the control room.
- As the UNSO manipulates the controls, all Control Rod position information on Panel 901-5 is lost.

Following initiation of the manual Reactor SCRAM, the UNSO is expected to ensure that the Mode Switch is in the ___(1)___ position.

With no Rod Position Indications on Panel 901-5 available, the shift could verify Control Rod positions by ___(2)___.

A. (1) RUN
(2) checking rod positions on POWERPLEX

B. (1) SHUTDOWN
(2) checking rod positions on POWERPLEX

C. (1) RUN
(2) obtaining resistance readings from the “Full In” and “00” PIP probe reed switches

D. (1) SHUTDOWN
(2) obtaining resistance readings from the “Full In” and “00” PIP probe reed switches

Answer: C

Explanation:

A) Incorrect. QOA 0010-05 directs the Mode Switch be left in Run. POWERPLEX uses control rod positions from the Rod Monitoring Program from the Plant Process Computer. However, only the last known good position is displayed.
B) Incorrect. QOA 0010-05 does not direct the Mode Switch to be placed in Shutdown. POWERPLEX uses control rod positions from the Rod Monitoring Program from the Plant Process Computer. However, only the last known good position is displayed.
C) Correct. QOA 0010-05 directs the Mode Switch be left in Run. QCOA 0280-01 RPIS Failure step D.7.c.(1).(d) directs obtaining resistance readings from the “Full In” and “00” PIP probe reed switches to obtain rod positions.
D) Incorrect. QOA 0010-05 does not direct the Mode Switch to be placed in Shutdown. QCOA 0280-01 RPIS Failure step D.7.c.(1).(d) directs obtaining resistance readings from the “Full In” and “00” PIP probe reed switches to obtain rod positions.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295016AA1.03	
Importance Rating	3.0	

K/A Statement: Control Room Abandonment: Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: RPIS (CFR: 41.7 / 45.6)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. LN-EVAC Control Room Evacuation, Rev. 5
2. QOA 0010-05 Plant Operation with Control Room Inaccessible, Rev. 26
3. QGA-101 RPV Control (ATWS) Rev. 15
4. QCOA 0280-01 RPIS Failure, Rev. 17
5. LIC-0280 Reactor Manual Control and Rod Position Information Systems, Rev. 14

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0280-K26 Given various plant conditions, PREDICT how Reactor Manual Control System (RMCS)/ Rod Position Information System (RPIS) and supported plant systems will respond to the following Reactor Manual Control System (RMCS)/ Rod Position Information System (RPIS) component or controller failures:
b. Loss of multiple RPIS input signals

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
Level of Difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:

Question: 7

Unit 1 is operating at 100% power.

- Annunciator 912-1 D-1, REACTOR BUILDING COOLING WATER LOW PRESSURE is in alarm.
- Unit 1 RBCCW Discharge Header Pressure 1-3740-4 indicates 37 psig and lowering
- Water is spraying from the RBCCW pipe on the return side of the online 1A Fuel Pool Cooling heat exchanger

RBCCW Header Isolation valve 1-3701 is then closed per QCOA 3700-01, RBCCW LOW PRESSURE.

- RBCCW Discharge Header Pressure stabilizes at approximately 40 psig.
- Annunciator 912-1 D-1, REACTOR BUILDING COOLING WATER LOW PRESSURE has NOT reset.

After several minutes, control room Operators note that 1A and 1B Reactor Recirculation pump seal cooling water temperatures are 118ºF and slowly rising.

Which of the following actions will stop the rise in recirculation pump seal temperatures?

A. Manually isolate the RBCCW return from the 1A Fuel Pool Cooling heat exchanger to the RBCCW return header.

B. Vent the running 1A and 1B RBCCW pumps and heat exchangers to remove air that entered the system from the leak location.

C. Declutch and manually throttle open 1-3701 to partially restore RBCCW system flow.

D. Align the ½ RBCCW pump and ½ RBCCW heat exchanger to Unit 1 to provide additional RBCCW cooling

Answer: A

Explanation:

A) Correct. RBCCW containment load return water will still be affected by the leak causing degraded system pressure, inventory, and cooling capacity. Individual loads should be isolated as possible to maximize the integrity and capability of the RBCCW system.

B) Incorrect. Plausible if the examinee concludes that air entrainment from the leak has bound pumps and heat exchangers preventing them from performing at their full capacity. Air should not enter the system as long as the system remains at a positive pressure.

C) Incorrect. Plausible if the examinee concludes that shutting the valve reduced flow to the RR pump seal coolers. The examinee may assume that restoring flow to the isolated outside of containment leg of RBCCW will assist in greater cooling for the RR pump seals.

D) Incorrect. Plausible if the examinee concludes that adding an additional pump and heat exchanger will assist in increasing the performance of the RBCCW system.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295018 2.4.35	
Importance Rating	3.8	

K/A Statement: PARTIAL OR COMPLETE LOSS OF CCW: Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects (CFR 41.10)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QCOA 3700-01, RBCCW Low Pressure, Rev. 10
2. QCOP 3700-02, RBCCW System Startup and Operation, Rev 29
3. QCAN 912-1, D-1, RX Building Cooling Water Low Pressure, Rev 01

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-3700-K26

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge ___
Comprehension or Analysis X

Level of difficulty: (1-5) 3

10 CFR Part 55 Content:
55.41 10
55.43 ______

Comments:

Question: 8

Given the following conditions:

- Unit 1 and 2 are at full power.
- 1A and U-2 Instrument Air Compressors (IAC) are running.
- Unit 1 Instrument Air Header pressure is 105 psig and lowering slowly
- Unit 2 Instrument Air Header pressure is 78 psig and lowering slowly.

An operator has been dispatched to start and align 1/2B IAC. Timely completion of this action will be necessary to prevent ________.

A. Feedwater Regulating Valves (FRVs) from failing closed

B. Control rods from drifting into the core

C. Loss of the ability to manually operate Safety Relief Valves (SRVs)

D. Instrument Air to Service Air "Little Joe" valves from opening

Answer: B

Explanation:

A) Incorrect. The FRVs lock up on a loss of air pressure.
B) Correct. A loss of IA pressure would cause a loss of scram air pressure allowing rods to drift into the core.
C) Incorrect. SRVs utilize Drywell Pneumatics (Nitrogen) for normal operation; IA can be manually aligned as a backup source.
D) Incorrect. The Unit 2 IA to Unit 2 Service cross-tie "Little Joe" valves would already be fully open (88 psig begin to open and fully open at 82 psig).

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295019AK3.02	
Importance Rating	3.5	

K/A Statement: Partial or Complete Loss of Instrument Air: Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Standby air compressor operation (CFR: 41.5 / 45.6)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QOA 4700-02, Instrument Air Compressor Trip, Rev. 18
2. QOA 4700-06, Loss of Instrument Air, Rev. 26

Proposed references to be provided to applicants during examination: None

Learning Objective: SRN-4701-K24 (as available)

Question Source:
Bank # ____
Modified Bank # X (Note changes or attach parent)
New ____

Question History: Last NRC Exam Dresden 2015 #4

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

Level of Difficulty: (1-5) 3

10 CFR Part 55 Content:
55.41 5
55.43 ____

Comments:
Changed answers B and C so that MSIVs is no longer the correct answer. Changed answer D to be Quad Cities specific.

Question: 9

Unit 2 is in day 14 of a planned 16 day refueling outage

- Reactor coolant temperature is 150ºF
- The reactor vessel head is fully tensioned
- RHR Pumps ‘A’ and ‘B’ are shutdown due the performance of QCOS 1000-28, “RHR Service Water Pump Comprehensive/Performance Test.”
- RHR Pump ‘C’ is out of service for motor replacement
- RHR Pump ‘D’ is operating in the Shutdown Cooling mode.

During the performance of QCOS 1000-28, MO 2-1001-5A, RHR HX SW DISCH VLV would not reopen following the OPEN/CLOSE exercise of the valve. Attempts to manually open the valve were also unsuccessful.

Which of the following identifies the Technical Specification LCO(s), if any, NOT being met?

A. All applicable Technical Specification LCOs are being met.

B. 3.4.8, Residual Heat Removal (RHR) Shutdown Cooling System-Cold Shutdown ONLY

C. 3.7.1, Residual Heat Removal Service Water (RHRSW) System ONLY

D. 3.4.8, Residual Heat Removal (RHR) Shutdown Cooling System-Cold Shutdown AND 3.7.1, Residual Heat Removal Service Water (RHRSW) System

Answer: B

Explanation:

A) Incorrect. Only one SDC subsystem is available. Plausible since RHR loop A was available for SDC until MO 2-1001-5A failed to open. Technical Specification LCO 3.4.8 permits one SDC subsystem to be inoperable for up to two hours for surveillances.
B) Correct. In Mode 4, Technical Specification 3.4.8 requires two shutdown cooling subsystems to be operable. Due to common components, both RHR and RHR service water pumps in one loop or one RHR and RHR service water in each loop along with a heat exchanger and the piping and valves to support must be operable. In this instance, there are no operable RHR service water pumps available in one loop while the other loop has only 1 RHR pump.
C) Incorrect. Tech Spec LCO 3.7.1 is only applicable in Modes 1, 2, and 3. Plausible since RHR loop A was available for SDC until MO 2-1001-5A failed to open. Additionally, RHRSW is required to support RHR SDC operation, but operability of RHRSW is covered by LCO 3.4.8 when in Mode 4 and 5.
D) Incorrect. Tech Spec LCO 3.7.1 is only applicable in Modes 1, 2, and 3. Plausible since RHRSW is required to support RHR SDC operation, but operability of RHRSW is covered by LCO 3.4.8 when in Mode 4 and 5.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295021 2.2.42	
Importance Rating	3.9	

K/A Statement: LOSS OF SHUTDOWN COOLING: Ability to recognize system parameters that are entry-level conditions for Technical Specifications (CFR 41.7, 41.10)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QCOP 1000-05, Shutdown Cooling Operation, Rev. 54
2. Tech Spec LCO 3.4.8, Residual Heat Removal (RHR) Shutdown Cooling System-Cold Shutdown
3. Tech Spec LCO 3.7.1, Residual Heat Removal Service Water (RHRSW) System

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-1000-K29, Given RHR/RHRSW system key parameter indications and various plant conditions, DETERMINE, from memory, if the RHR/RHRSW system Tech Spec LCOs have been met.

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	___
	Comprehension or Analysis	x
Level of difficulty:	(1-5) 4	
10 CFR Part 55 Content:	55.41 7,10	
	55.43 _____	

Comments:

Question: 10

Unit 1 is in a refueling outage. Core reload is in progress. You are the Unit 1 Reactor Operator.

- An irradiated fuel bundle is being moved from the Spent Fuel Pool to the reactor cavity.
- As the fuel bundle is being lowered into a peripheral core location, the bottom of the fuel bundle contacts the core shroud and becomes ungrappled.
- The fuel bundle comes to rest between the reactor pressure vessel wall and the core shroud.
- The fuel bundle integrity appears to be maintained.

Which one of the following workers is at the greatest risk of radiation overexposure?

A. Instrument Technician working at the SBLC Tank

B. Refueling SRO on the bridge

C. Mechanic working on SRVs

D. RP Tech at Refuel Floor Access Point

Answer: C

Explanation:

A) Incorrect: SLC Tank is in Secondary Containment, and exposure is limited by shielding from the water in the RPV and cavity, Primary Containment, as well as distance from the core.
B) Incorrect. SRO on the bridge is shielded by water level within the RPV and cavity,
C) Correct. The mechanic is closest to the irradiated fuel and has the least amount of shielding between the reactor and his current location.
D) Incorrect. The RP Tech's exposure is limited by shielding from the water in the RPV and cavity, Primary Containment, as well as distance from the core.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295023AK1.01	
Importance Rating	3.6	

K/A Statement: Refueling Accidents: Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS: Radiation exposure hazards (CFR: 41.8 to 41.10)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
1. QCFHP 0110-02 Inadvertent Criticality During Fuel Moves, Rev. 4

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (as available)

Question Source:
Bank # X
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam Pilgrim 2013 question #39

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

Level of Difficulty: (1-5) 2

10 CFR Part 55 Content:
55.41 9
55.43 ____

Comments:

Question: 11

Unit 1 experienced a LOCA.

- All control rods are fully inserted.
- Reactor water level is -100 inches and steady.
- Reactor pressure is 70 psig and steady.
- Drywell pressure is 10 psig and steady.
- Torus temperature is 150ºF and rising.
- Neither Core Spray pump is available.
- The ‘A’ RHR loop is operating in the Drywell Spray mode.
- The ‘B’ RHR loop is operating in the LPCI injection mode.

In accordance with QGAs and QCOP 1000-30, “Post-Accident RHR Operation,” which of the following actions may be performed to further lower Drywell Pressure?

A. Establish Drywell Sprays on the RHR B Loop.

B. Fully open MO 1-1001-36A, TORUS H2O TEST VLV; maximize RHRSW flow using pumps A and B;

C. Ensure that 1-1001-16A, RHR HX BYP VLV is full OPEN.

D. Ensure that 1-1001-16A, RHR HX BYP VLV is full SHUT; maximize RHRSW flow using pumps A and B.

Answer: D

Explanation:

In order for Drywell Sprays to be more effective the Torus must be cooled down. Heat removal from the Torus can be maximized by maximizing RHR and RHRSW flow through ‘A’ RHR Heat Exchanger (RHR Pumps A and B, RHRSW Pumps A and B, and ensuring the Hx is NOT bypassed)

A) Incorrect. Plausible if the examinee fails to recognize that LPCI injection from the RHR ‘B’ loop is necessary for adequate core cooling; if flow is diverted RPV level will lower.
B) Incorrect. While this action may increase torus cooling, this action would divert flow from RPV injection and jeopardize core cooling.
C) Incorrect. While this action may increase RHR system flow it will reduce heat removal capability.
D) Correct. QGA 200 instructs operators to maximize torus cooling using available RHR pumps with consideration for ensuring adequate core cooling. Maximization of Torus Cooling would include the actions specified.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	______
	Group #	1	______
	K/A #	295024EA2.06	
	Importance Rating	4.1	

K/A Statement: HIGH DRYWELL PRESSURE: Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: Suppression pool temperature (CFR 41.10)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	1. QGA 200, Primary Containment Control, Rev. 11

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0001-K23

Question Source: Bank # _______
Modified Bank # _______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge _
Comprehension or Analysis X

Level of difficulty: (1-5) 3

10 CFR Part 55 Content: 55.41 10
55.43 _____

Comments:

Question: 12

A manual SCRAM was initiated on Unit 1 due to a rupture in the Reactor Building Instrument Air (IA) Header.

- All control rods are fully inserted.
- QGA 100 is being implemented.
- Reactor pressure is cycling on the ERVs.
- The Reactor Building IA Header has been isolated.

Which of the following describes how the Main Turbine Bypass Valves could be used to control RPV pressure?

Main Turbine Bypass Valves may be used …

A. independently of the Main Steam Line Drain Valves.

B. in conjunction with the Main Steam Line Outboard Drain Valves alone.

C. in conjunction with the Main Steam Line Inboard Drain Valves alone.

D. in conjunction with both the Main Steam Line Inboard and Outboard Drain Valves.

Answer: D

Explanation:

A) Incorrect. The outboard MSIVs are shut as a result of the isolation of the Instrument Air line rupture. Unless the Main Steam Lines are re-pressurized the main turbine bypass valves are unavailable. Re-pressurization requires use of both the Inboard and Outboard drain lines. Plausible because this would be a viable response with the inboard MSIVs open.
B) Incorrect. A flow-path from the RPV will not exist unless the Inboard and Outboard Drain Lines are cross-tied via MO 1-220-3.
C) Incorrect. A flow-path from the RPV will not exist unless the Inboard and Outboard Drain Lines are cross-tied via MO 1-220-3.
D) Correct. The Main Steam Line Inboard and Outboard Drain lines must be cross-tied via MO 1-220-3 to establish a flow-path from the RPV to the Main Turbine Bypass Valves.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295025EA1.01	
Importance Rating	2.9	

K/A Statement: High Reactor Pressure: Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: Main steam line drains (CFR: 41.7 / 45.6)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. L-QGA100 QGA 100, RPV Control, Rev. 11
2. QGA 100, RPV Control, Rev. 11
3. QCOP 0250-05 Reactor Pressure Control Using Main Steam Line Drains, Rev. 6
4. LIC-0250 Main Steam, Rev. 18

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0001-K17 Given QGA 100, "RPV Control", EXPLAIN the reasons for the actions.

Question Source:

Bank # _______
Modified Bank # _______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	X
Level of Difficulty:	(1-5) 3	
10 CFR Part 55 Content:	55.41 7	
	55.43 ______	

Comments:

Question: 13

Unit 2 has experienced a LOCA.

- Torus Bottom pressure is 6 psig
- Suppression Pool level is 14.5 feet
- Drywell Pressure is 9 psig
- Torus Water Temperature is 180°F
- BOTH RHR Loop A pumps are in Torus/Drywell Spray mode with total RHR Loop A flow of 9000 gpm (assume the pumps are sharing the load equally)
- 2A Core Spray pump flow is 4400 gpm
- NO other ECCS Pumps, nor RCIC are running

Which of the following ECCS pumps, if any, have sufficient NPSH to avoid cavitation. [NPSH Curves provided.}

A. NONE

B. RHR Loop A pumps ONLY

C. 2A Core Spray pump ONLY

D. RHR Loop A pumps AND the 2A Core Spray pump

Answer: C

Explanation:

A) Incorrect. The 2A Core Spray pump is operating with adequate NPSH (Attachment V). Plausible, since examinee could reach this conclusion if they use the wrong curve (Attachment S, T, or U).
B) Incorrect. Plausible if the DW pressure is used to read the RHR pump curve.
C) Correct. Per Attachment V, the 2A Core Spray pump would fall below the 6 psig curve at 4400 gpm and RHR pumps at 4500 gpm each would be above the 6 psig curve at Torus temperature of 180°F.
D) Incorrect. 2A Core Spray is operating with adequate NPSH, RHR Loop A pumps are not.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	______
	Group #	1	______
	K/A #	295026EK2.02	
	Importance Rating	3.6	

K/A Statement: SUPPRESSION POOL HIGH WATER TEMPERATURE: Knowledge of the interrelations between SUPPRESSION POOL HIGH WATER TEMPERATURE and the following: Suppression pool spray (CFR 41.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
1. QCAP 0200-10, Emergency Operating Procedure (QGA) Execution Standards; Rev 54
2. L-QGADET; Rev 10

Proposed references to be provided to applicants during examination:
1. QCAP 0200-10, Attachments R through W; RHR and CS NPSH CURVES

Learning Objective: SR-0001-K09p

Question Source: Bank # ______
Modified Bank # X (Parent Attached)
New _

Question History: Last NRC Exam Browns Ferry NRC ILT Exam 2011

Question Cognitive Level: Memory or Fundamental Knowledge _
Comprehension or Analysis X

Level of difficulty: (1-5) 2

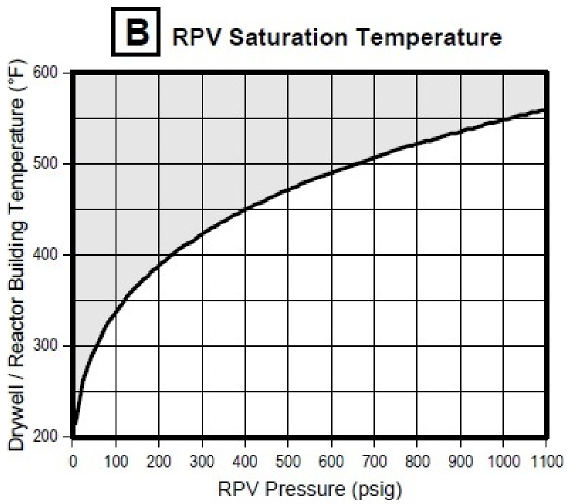
10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments:
Stem parameters changed resulting in a different correct answer.

Question: 14

Unit 2 was operating at near rated power, when a LOCA occurred.

Reactor pressure is 100 psig and LOWERING slowly.



C RPV Level Instrument Criteria

Instrument	Range (in.)	Use only if...
Fuel Zone	–340 to +60	Indicated level above –299 in.
Lower Wide Range	–344 to +66	Indicated level above –299 in.
Medium Range	–60 to +60	Indicated level above –42 in. OR Reactor building temperature below 185°F
Upper Wide Range	–42 to +358	Indicated level above 70 in.
Narrow Range	0 to +60	Indicated level on-scale.

Which one of the following sets of parameters [(1) Drywell Temperature; (2) Lower Wide Range RPV Water Level], would result in a USABLE Lower Wide Range RPV Water Level indication?

A. (1) 250 degrees F
(2) -270 inches

B. (1) 290 degrees F
(2) -340 inches

C. (1) 330 degrees F
(2) -299 inches

D. (1) 370 degrees F
(2) -305 inches

Answer: A

Explanation:

Lower wide range RPV water level instrument range is -344 inches to +66 inches however it is only usable if above -299 inches. If above saturation temperature, water level instruments may be unreliable. At 100 psig reactor pressure, saturation temperature is approximately 340 degrees F. At 250 degrees F and indicated level at -270 inches, the Lower Wide Range reactor water level instruments would still be usable.

It is important for operators to be able to determine whether or not the RPV water level instruments are reading correctly. The operational implications of not having reliable level indications are to flood the RPV. All distractors are plausible based on interpreting both the graph of RPV pressure vs Drywell temperature and the chart that determines the criteria for usability.

Distractor B is incorrect: Below Saturation Temperature but NOT ABOVE Minimum Usable level of -299 inches.
Distractor C is incorrect: Below Saturation Temperature but NOT ABOVE Minimum Usable level of -299 inches.
Distractor D is incorrect: Above Saturation Temperature and NOT ABOVE Minimum Usable level of -299 inches.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295028EK1.01	
Importance Rating	3.5	

K/A Statement: High Drywell Temperature: Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Reactor water level measurement (CFR: 41.8 to 41.10)

Technical Reference(s): QGA 100 Rev 11
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0263-K22 (as available)

Question Source: Bank # X
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam Quad Cities 2014 #14

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	X
Level of Difficulty:	(1-5) 3	
10 CFR Part 55 Content:	55.41 8 to 10	
	55.43 _____	

Comments:

Replaced the RPV Water Level indication for choice "C" with a value within the instrument range of indication; original value was outside of the range of indication.

Question: 15

Unit 1 is in Mode 4, and the following plant conditions exist:

- Torus level is 7.0 feet and steady following leak repairs
- A/B CCSTs level is 18 feet ('A' and 'B' CCSTs are cross tied)
- Hotwell level is 10 inches
- Floor Drain Surge Tank level is 20% with acceptable chemistry conditions

Which of the following procedures provides the quickest method to transfer the LARGE quantity of water necessary to restore the Torus water level?

A. QCOP 1000-19, MAIN CONDENSER WATER TRANSFER TO THE TORUS

B. QCOP 1000-28, TORUS FILLING FROM THE CCST WITH BOTH RHR SUCTION VALVES OPEN

C. QCOP 1600-16, FLOOR DRAIN SURGE TANK TRANSFER TO TORUS

D. QCOP 1300-03, FILLING TORUS FROM THE CCST THROUGH THE RCIC MINIMUM FLOW LINE

Answer: B

Explanation:

A) Incorrect. Plausible as transfer of water to the Torus from the main condenser is considered the best option for small transfers of water per QCOP 1600-12. Note in this instance a large transfer is needed and main condenser level is only 10 inches which is when operators must start to monitor for a loss of NPSH to the condensate pumps.
B) Correct. Transfer from the CCST to the torus via the RHR suction lines would be the most appropriate procedure for plant conditions for a large inventory move.
C) Incorrect. Plausible as transfer from the Floor Drain Surge Tank with acceptable chemistry is an option for a small inventory shift to the Torus per QCOP 1600-16. Also starting level in the FDST will result in FDST transfer pumps tripping on low level before adequate inventory can be shifted.
D) Incorrect. Plausible as this is an acceptable procedure for small inventory shifts due to the smaller piping used.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295030EA1.06	
Importance Rating	3.4	

K/A Statement: LOW SUPPRESSION POOL WATER LEVEL: Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Condensate storage and transfer (makeup to the suppression pool) (CFR 41.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.

1. QCOP 1600-12, Torus Normal Level Control, Fill and Drain Procedure Directory
2. QCOP 1000-28, Torus Filling from the CCST with Both RHR Suction Valves Open, Rev. 12
3. QCOP 1600-16, Floor Drain Surge Tank Transfer to the Torus, Rev. 12

Proposed references to be provided to applicants during examination: None

Learning Objective: SRN-1601-K19

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	___
	Comprehension or Analysis	_X_
Level of difficulty:	(1-5) __3__	
10 CFR Part 55 Content:	55.41 __7__	
	55.43 _____	

Comments:

Question: 16

QGA-101 RPV Control (ATWS), has directed preventing injection except from Boron, CRD, and RCIC and lowering Reactor Water Level to below –35 inches.

Which of the following describes an operational consequence of performing these steps?

A. Improved boron mixing

B. Increased core inlet sub-cooling

C. Minimized neutron flux oscillations

D. Increased heat addition to the Suppression Pool

Answer: C

Explanation:

A) Incorrect. Lowering RPV level reduces natural circulation (Note: Recirc Pumps would be tripped during ATWS actions) which reduces Boron mixing. Plausible because –The actions to control RPV water level in QGA-101, differ from those in the Level control leg of QGA-100 to address four basic concerns: The first concern includes, promote boron mixing however this is accomplished by override #8 after hot shutdown boron injection is reached by raising Reactor Water level.
B) Incorrect. Lowering RPV Water level below –35 inches places the feedwater spargers in the steam space providing effective heating of the relatively cold feedwater and eliminating the potential for high core inlet subcooling. Plausible because – Core Inlet subcooling is affected by changing RPV water level. Additionally RCIC and CRD enter the Feedwater line downstream of the Feedwater Heaters and the candidate may think this cooler water is the overriding effect.
C) Correct. To prevent or mitigate the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities, RPV water level is lowered.
D) Incorrect. QGA-101 controls RPV water level to minimize suppression pool heatup. Twenty-four inches below the lowest nozzle in the feedwater sparger (-35 inches) has been selected as the upper bound of the RPV water level control band. This water level is sufficiently high that control of RPV water level with feedwater pumps can preclude the MSIV low water level isolation. Plausible because – While intentionally lowering level –35 inches is the top of the RPV water level band and the bottom of the band has not yet been given. The candidate may assume that the MSIVs will close on low level which would result in increased heat addition to the Suppression Pool.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295031EK1.03	
Importance Rating	3.7	

K/A Statement: Reactor Low Water Level: Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: Water level effects on reactor power (CFR: 41.8 to 41.10)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	1. QGA 101, RPV CONTROL (ATWS), Rev. 15

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0001-K061 Explain the reasons for actions (as available)

Question Source: Bank # X
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History:	Last NRC Exam	Browns Ferry 2017 #21	
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
Level of Difficulty:	(1-5)	3	
10 CFR Part 55 Content:	55.41	5	
	55.43		

Comments:

Updated question to match Quad Cities specific procedures and numbers.

Question: 17

A hydraulic ATWS has occurred on Unit 2:

- RPV pressure is being controlled with the ERVs
- PCIS Group I, II, III isolations have occurred
- Torus Bulk Temperature is 94°F and rising 1°F/min
- SBLC is NOT injecting

When is SBLC injection FIRST REQUIRED to preclude HCTL from being exceeded?

A. 16 minutes

B. 27 minutes

C. 70 minutes

D. 77 minutes

Answer: A

Explanation:

A) Correct. When torus temperature reaches 110°F, SBLC must be injected to allow hot shutdown boron weight to be injected before HCTL is reached.
B) Incorrect. Plausible as this corresponds to TS LCO 3.6.2.1 Condition E (exceeding 120°F bulk torus temperature) which would require depressurizing the RPV to < 150 psig within 12 hours.
C) Incorrect. Plausible as this corresponds to the HCTL of 164°F bulk torus temperature at 1000 psig (upper limit of the expected pressure control band.
D) Incorrect. Plausible as this corresponds to the HCTL of 171°F bulk torus temperature at 800 psig (lower limit of the expected pressure control band.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295037EA2.07	
Importance Rating	4.0	

K/A Statement: SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: containment conditions/isolations (CFR 41.10)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	1. QGA 200, Primary Containment Control, Rev. 23 2. QGA 101, RPV Control (ATWS), Rev. 15

Proposed references to be provided to applicants during examination:

HCTL Curve (QGA-200, Figure M)

Learning Objective: SR-0001-K61

Question Source:
Bank # X
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam Dresden NRC ILT 2017

Question Cognitive Level:	Memory or Fundamental Knowledge	_X_
	Comprehension or Analysis	____
Level of difficulty:	(1-5) __2__	
10 CFR Part 55 Content:	55.41 __10__	
	55.43 _____	

Comments: Added HCTL Curve and modified the distractors to test the examinees knowledge of applicability of the HTCL Curve during an ATWS (curve assumes that the reactor is shutdown).

Question: 18

QGA 400, RADIOACTIVITY RELEASE CONTROL, directs: “Isolate all primary system discharges outside primary and secondary containments except systems needed for other QGA actions.”

Why are systems required for other QGA actions specifically exempted?

A. These systems have engineering and administrative controls in place to minimize leakage of highly radioactive fluids during an accident.

B. Any leakage from these systems will be collected, processed, and released from monitored paths.

C. Any leakage from these systems will have very low activity levels and will NOT significantly contribute to the overall radiological consequences of the event.

D. Isolation of these systems could ultimately result in a much larger radiological release.

Answer: D

Explanation:

A) Incorrect: Plausible because TS 5.5.2 requires the facility to have a program in place to control leakage from systems outside the primary containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. However not all the systems that may be exempted, fall into this category.
B) Incorrect: Plausible because liquid leakage is normally collected by floor and equipment drain systems and gaseous leakage within the secondary containment would be collected by the Standby Gas Treatment System before being released to the exhaust stack which is monitored. However, if leakage is large, or if it bypasses the collection systems, it could significantly contribute to off-site doses.
C) Incorrect: Plausible because many, but not all, of the exempted systems would have low activity levels.
D) Correct: Systems exempted from isolation are necessary for other QGA actions. EPG basis indicates that their isolation may cause degradation of plant parameters and result in an increase in offsite release.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295038EK3.02	
Importance Rating	3.9	

K/A Statement: High Off-Site Release Rate: Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: System isolations (CFR: 41.5 / 45.6)

Technical Reference(s): QGA 400, Radioactivity Release Control, Rev. 9
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0001-K35 Given QGA 400, “Radioactivity Release Control”, EXPLAIN the reasons for the actions.

Question Source:
Bank # X
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam None

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
Level of Difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	5
	55.43	

Comments:

Question: 19

The Unit 1 HPCI pump is started for routine surveillance testing.

Approximately 5 minutes later, the HPCI auxiliary oil pump motor windings fail resulting in significant arcing and flames protruding from the motor casing.

Before the fire can be extinguished the HPCI room Protectowire system is actuated from the high temperature of the flames.

How will the fire protection system respond this event?

A. After a short delay, there will be a timed discharge of CO_2 into the HPCI room.

B. The HPCI room's fire protection deluge (spray) system actuates.

Both diesel driven fire pumps will start.

C. The sprinklers will actuate only if temperatures at the sprinkler heads melts/ruptures the sprinkler head fusible link/bulb.

Both diesel driven fire pumps will start if the wet pipe sprinklers initiate.

D. The HPCI room's dry pipe sprinkler system will fill with water.

The sprinklers will actuate only if temperatures at the sprinkler heads melts/ruptures the sprinkler head fusible link/bulb.

Both diesel driven fire pumps will start.

Answer: B

Explanation:

A) Incorrect. Plausible if the HPCI rooms were protected by the automatic CARDOX system.
B) Correct. HPCI rooms are protected by a deluge system which is automatically initiated by a Protectowire system when its temperature exceeds 190°F or by local manual actuation.
C) Incorrect. Plausible if the HPCI rooms were protected by a wet pipe sprinkler system.
D) Incorrect. Plausible if the HPCI rooms were protected by a dry pipe fire pre-action system.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	600000AK2.01	
Importance Rating	2.6	

K/A Statement: PLANT FIRE ON SITE: Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Sensors/detectors and valves

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QCOA 0010-12, Fire Explosion, Rev. 47
2. QCOP 4100-04, Resetting Grinnell Multimatic And Tyco External Resetting Valves, Rev. 12

Proposed references to be provided to applicants during examination: None

Learning Objective: SRN-4100-K07 and SR-4100-K14

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ____

Level of difficulty: (1-5) 3

10 CFR Part 55 Content:
55.41 7
55.43 _____

Comments:

Question: 20

Both Units are at full power when a Grid Disturbance occurs affecting both Units.

The Unit 1 ANSO reports:

- System Frequency indication on the 912-2 panel shows 60.5 Hz and is slowly RISING
- Annunciator 901-7 G-3 TURBINE BYPASS VALVE OPEN is in alarm.

If grid frequency continues its slow rise, which of the following conditions will be the first to cause the Units to SCRAM?

A. The CVs and IVs fast closing.

B. CVs throttling closed.

C. RPS MG Sets EPA's trip.

D. Turbine trips on over-speed.

Answer: B

Explanation:

The result of grid frequency raising would be an increase in speed on the Main Generator. Control valves control turbine speed during overspeed conditions. As the CVs close the Turbine Bypass Valves will open until all are full open. Once the Bypass Valves are full open and grid frequency continued to rise, reactor pressure would increase to the point of a high pressure scram.

A) Incorrect: Plausible because during a Power Load Unbalance condition the CVs and IVs fast close together.
B) Correct.
C) Incorrect: Plausible because the EPAs have an under frequency, under voltage, and over voltage trip. There is no over frequency trip.
D) Incorrect: Plausible because the turbine would trip when frequency reached 66 Hz (110% turbine speed). However the reactor would have scrammed on high pressure before reaching 66 Hz (approximately 61.5 Hz).

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	700000AA2.06	
Importance Rating	3.4	

K/A Statement: Generator Voltage and Electric Grid Disturbances: Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Generator frequency limitations (CFR: 41.5 and 43.5 / 45.5, 45.7, and 45.8)

Technical Reference(s): (Attach if not previously provided, including version/revision number.

1. QCOA 6000-02, Main Generator Abnormal Operation; Rev. 20
2. QC UFSAR Section 10.2.2; Rev 14 (October 2017)
3. LIC 5652a, DEHC Electrical; Rev 7

Proposed references to be provided to applicants during examination: None

Learning Objective: LIC 5652a-K20, Given a Main Turbine Control - EHC Logic System operating mode and various plant conditions, EVALUATE the following Main Turbine Control - EHC Logic System indications/responses and DETERMINE if the indication/ response is expected and normal: c. Turbine speed

Question Source:
Bank # ________
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	____
	Comprehension or Analysis	X
Level of Difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	5
	55.43	____

Comments:

Question: 21

A reactor startup from cold shutdown is in progress on Unit 2.

The reactor is critical with a 320 second period.

The next control rod is withdrawn from position 08 to 10 resulting in a sustained 40 second period.

What is the NEXT required operator action?

A. Monitor overlap data between SRMs and IRMs and range IRMs as necessary

B. Position SRM detectors as necessary to maintain count rate between 10^3 and 10^5 cps

C. Inform the Shift Manager and the Qualified Nuclear Engineer of the power rise, then insert the Control Rod as far as necessary to turn power

D. Insert the Control Rod back to position 08 to obtain a reactor period of > 50 Seconds, then notify the Shift Manager and Qualified Nuclear Engineer

Answer: D

Explanation:

A) Incorrect. Plausible as monitoring the SRM/IRM overlap verification prior to removing SRMs is a required action per QCGP 1-2, “Normal Unit 2 Start Up” but operating with a period less than 50 seconds is not permitted.
B) Incorrect. Plausible because SRM detectors will be withdrawn to prevent an upscale rod block, but SRM withdrawal will not correct the short period and withdrawal could mask the short period condition.
C) Incorrect. Plausible because the SM/QNE are required to be informed, but not until after the control rod is inserted. Additionally this is the action that would be taken if the reactor went critical outside of the estimated critical position (± 1% ECP error).
D) Correct. Action required per the ARP for an SRM short period.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295014AA1.03	
Importance Rating	3.5	

K/A Statement: INADVERTANT REACTIVITY ADDITION: Ability to operate and/or monitor the following as they apply to INADVERTANT REACTIVITY ADDITION: RMCS (CFR 41.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	1. QCAN 902-5, E-5, SRM Short Period, Rev. 06 2. QCGP 1-2, Normal Unit 2 Start Up, Rev. 29

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-RXTH-K24

Question Source: Bank # X
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam Fermi NRC ILT Exam 2015

Question Cognitive Level: Memory or Fundamental Knowledge ___
Comprehension or Analysis X

Level of difficulty: (1-5) 3

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Question: 22

Unit 2 is at 100% power.

- An instrument failure has resulted in a trip of RPS Channel B.
- HALF of the control rods scram.

Which one of the following describes the expected status of the RPS SCRAM SOLENOID GROUP indicating lights?

A. HALF of the Channel A RPS SCRAM SOLENOID GROUP indicating lights DARK.
ALL Channel B RPS SCRAM SOLENOID GROUP indicating lights DARK.

B. HALF of the Channel A RPS SCRAM SOLENOID GROUP indicating lights DARK.
HALF of the Channel B RPS SCRAM SOLENOID GROUP indicating lights DARK.

C. ALL Channel A RPS SCRAM SOLENOID GROUP indicating lights LIT.
ALL Channel B RPS SCRAM SOLENOID GROUP indicating lights DARK.

D. ALL Channel A RPS SCRAM SOLENOID GROUP indicating lights DARK.
ALL Channel B RPS SCRAM SOLENOID GROUP indicating lights DARK.

Answer: A

Explanation:

A) Correct. A single Scram signal on Channel B would cause all of that channel's RPS SCRAM SOLENOID GROUP indicating lights to go dark. For half of the control rods to insert, one or more malfunctions (e.g., failure of a single actuator relay in logic leg A1 or A2; or blown fuses in two separate solenoid groups) had to occur resulting in the trip of two RPS Channel A Solenoid Groups concurrently with the trip signal inputted to Channel B RPS .
B) Incorrect. A Scram signal on Channel B would cause all of that channel's RPS SCRAM SOLENOID GROUP indicating lights to go dark.
C) Incorrect. This would be the expected indication with no control rods scramming during RPS testing.
D) Incorrect. If these were the indications on RPS, all of the rods should have inserted. Since the scram signal was only inserted into the B Channel, no control rods should have inserted.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295015AK2.04	
Importance Rating	4.0	

K/A Statement: INCOMPLETE SCRAM: Knowledge of the interrelations between INCOMPLETE SCRAM and the following: RPS (CFR: 41.7 / 45.8)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	1. QCOA 0300-16, Determination of Control Rod Motion Troubles Rev. 1 2. QCOA 0500-01, Partial Scram Actuation, Rev. 8

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0500-K26 Evaluate given key RPS parameter indications and/or responses depicting a system specific abnormality/failure to correct a partial half scram.

Question Source:
- Bank # ____
- Modified Bank # ____ (Note changes or attach parent)
- New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
- Memory or Fundamental Knowledge ____
- Comprehension or Analysis X

Level of Difficulty: (1-5) 3

10 CFR Part 55 Content:
- 55.41 7
- 55.43 ____

Comments:

Question: 23

Unit 1 is at 70% reactor power with a rod pattern adjustment in progress.

The running control rod drive pump experiences a shaft shear failure.

After the start of the standby CRD pump,

- The in-service flow control valve (FCV) would not reopen.
- The standby FCV is being placed in service.

The following control rod drive mechanisms temperatures are in alarm at TR 1-340-16:

- K-06 – 405°F
- F-09 – 400°F
- J-12 – 385°F
- A-09 – 350°F
- L-09 – 265°F

Which of the following describes: (1) the impact to the CRDM; and (2) the required action(s) once cooling flow is reestablished?

A. (1) Increased mechanical friction, due to differential thermal expansion between CRDM internals, will slow control rod movement.

(2) Allow CRDM temperatures to decrease below 350°F before moving control rods

B. (1) Leakage past the internal seals, due to seal embrittlement and eventual failure, will slow control rod movement.

(2) Allow CRDM temperatures to decrease below 350°F before moving control rods

C. (1) Increased mechanical friction, due to differential thermal expansion between CRDM internals, will slow control rod movement.

(2) Fully insert and disarm all control rods with drive temperatures having exceeded 350°F, until scram time testing can be performed.

D. (1) Leakage past the internal seals, due to seal embrittlement and eventual failure, will slow control rod movement.

(2) Fully insert and disarm all control rods with drive temperatures having exceeded 350°F, until scram time testing can be performed.

Answer: B

Explanation:

E) Incorrect. (1) Plausible as metals expand with temperature increases, however temperatures are well below any value that would be of concern. (2) Limiting control rod movement is the correct action for overheating of the CRDM seals.
F) Correct. Normal control rod movements with CRDM temperatures above 350°F can cause seal embrittlement and eventual failure.
G) Incorrect. (1) Plausible as metals expand with temperature increases, however temperatures are well below any value that would be of concern. (2) Plausible, since SCRAM times may be affected by overheating, which may affected control rod operability.
H) Incorrect. (1) Correct impact. (2) Plausible, since SCRAM times may be affected by overheating, which may affected control rod operability.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295022AA2.03	
Importance Rating	3.1	

K/A Statement: LOSS OF CRD PUMPS: Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS: CRD mechanism temperatures (CFR 41.10)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QCOA 0300-01, Control Rod Drive Pump Failure, Rev. 18
2. QCOS 0300-21, CRD Temperature Surveillance, Rev. 20

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0302-K22h

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge ___
Comprehension or Analysis X

Level of difficulty: (1-5) 3

10 CFR Part 55 Content:
55.41 10
55.43 _____

Comments:

Question: 24

A Loss of Coolant Accident inside the Drywell is in progress.

Which one of the following failures or conditions could result in exceeding NEGATIVE design pressure rating of the containment?

A. Torus to drywell vacuum breaker failing open.

B. Torus level rising to 18 feet with drywell sprays in service.

C. Torus level rising to 19 feet with torus sprays in service.

D. SRV tailpipe vacuum breaker failing closed.

Answer: B

Explanation:

A.) Incorrect. This event would challenge the over pressure rating of the containment as steam would bypass the suppression pool.
B.) Correct. At 17 feet the Torus to Drywell Vacuum Breakers begin to be covered. With drywell sprays in service, the vacuum breakers would be unable to relieve back to the drywell, resulting in the drywell going negative in pressure. QGA-200 Primary Containment Control directs that drywell sprays be secured at this level.
C.) Incorrect. 18.5 feet is the point at which the SRV Tail Pipe Level Limit becomes limiting. If the SRV Tail Pipe Level Limit failed, a potential loss of pressure suppression might occur which is a high pressure challenge to the containment.
D.) Incorrect. A SRV Tailpipe vacuum breaker failing closed would result in a vacuum drag of water up the tailpipe. Subsequent SRV lifts would result in large hydro dynamic forces on the tailpipe with possible failure. At most this would result in a high pressure condition if the tail pipe failed.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295029EK1.01	
Importance Rating	3.4	

K/A Statement: High Suppression Pool Water Level: Knowledge of the operational implications of the following concepts as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Containment integrity (CFR: 41.8 to 41.10)

Technical Reference(s): QGA-200, Primary Containment Control, Rev. 11
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (as available)

Question Source:
Bank # X
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam Pilgrim 2010 #59

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	X
Level of Difficulty:	(1-5)	______
10 CFR Part 55 Content:	55.41	5
	55.43	______

Comments:

Only changes to the question were to use Quad Cities specific values.

Question: 25

Unit 2 is operating at 100% power when an invalid FULL Group 2 isolation occurs.

Which of the following is most likely to cause entry into QGA 300?

A. ‘D’ Heater Bay High Temperature

B. Reactor Building Ventilation high radiation

C. HPCI Room area radiation

D. MSIV Room High Temperature

Answer: D

Explanation:

A) Incorrect. 'D' Heater Bay high temperature is not a QGA 300 entry condition. Plausible because the 'D' Heater Bay is immediately adjacent secondary containment and uses a common entrance with the MSIV room.
B) Incorrect. The cause of the Group 2 isolation was an invalid signal, therefore no accident conditions exist and therefore rad levels should be normal. Plausible because Reactor Building Vent high radiation is a QGA 300 entry condition.
C) Incorrect. The cause of the Group 2 isolation was an invalid signal, therefore no accident conditions exist and therefore rad levels should be normal. Plausible because HPCI Room Area high radiation is a QGA 300 entry condition.
D) Correct. QOA 5750 specifically cautions operators to restore Reactor Ventilation to normal as soon as possible in order to avoid an MSIV Room high temperature alarm condition.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295032EK2.02	
Importance Rating	3.6	

K/A Statement: HIGH SECONDARY CONTAINMENT AIR TEMPERATURE: Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AIR TEMPERATURE and the following: Secondary containment ventilation (CFR 41.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QOA 5750-07, Reactor Building Ventilation Isolation, Rev. 12
2. QGA 300, Secondary Containment Control, Rev. 13

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-5750-K24 and SR-0001-K29

Question Source:
Bank # X
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam Quad Cities NRC ILT 2002

Question Cognitive Level:
Memory or Fundamental Knowledge ___
Comprehension or Analysis X

Level of difficulty: (1-5) 2

10 CFR Part 55 Content:
55.41 7
55.43 ______

Comments: Changed answer choice 'A' to eliminate 2nd possible correct answer.

Question: 26

Unit 2 is at 100% power with the “Unit 2 Standby Gas Treatment Initiation and Reactor Building Ventilation Isolation Test” (QCOS 7500-08) in progress.

At several times during performance of the test, neither the Reactor Building Ventilation and nor Standby Gas Treatment systems are in operation

While neither system is in operation, Reactor Building D/P goes positive.

Which of the following describes the expected operator response to this condition?

A. QGA 300 entry is required;
Exit QCOS 7500-08 and restore Reactor Building Ventilation to service.

B. QGA 300 entry is NOT required; this is an anticipated condition;
Continue performance of QCOS 7500-08 actions.

C. QGA 300 entry is required;
Continue performance of QCOS 7500-08 actions in parallel with QGA 300.

D. QGA 300 entry is NOT required; however, this condition is unexpected
Exit QCOS 7500-08 and restore Reactor Building Ventilation to service.

Answer: C

Explanation:

A) Incorrect: The high d/p is an anticipated condition and will be remedied with continuation of the next step. Plausible because QGA 300 entry is required and restoring Reactor Building Ventilation would be the expected action if continuation of the test procedure did not correct the condition.
B) Incorrect: QGA 300 entry is required. Plausible because the procedure directs the operator to secure both Reactor Building Ventilation and STBTS, and starting both trains of SBGTS would restore Reactor Building D/P.
C) Correct: Even though the procedure directs the operators to secure both Reactor Building Ventilation and Standby Gas Treatment System, any time an entry condition is met for a QGA it must be entered. Continued performance of QCOS 7500-08 will direct the operators to remove fuses to cause the auto start of the standby train of Standby Gas.
D) Incorrect: QGA 300 entry is required. Plausible because exiting the test procedure should return the systems operation.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295035G2.2.12	
Importance Rating	3.7	

K/A Statement: Secondary Containment High Differential Pressure: Knowledge of surveillance procedures (CFR: 41.10 / 45.13)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QCOS 7500-08, “Unit 2 Standby Gas Treatment Initiation and Reactor Building Ventilation Isolation Test”; Rev 24
2. QGA 300, “Secondary Containment Control”; Rev 13

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-7500-K26; EVALUATE given key SBGTS 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): d. Low or High system differential pressure

Question Source:
Bank # ________
Modified Bank # ________ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	X
Level of Difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	10
	55.43	______

Comments:

Question: 27

Unit 1 is at 100% power, annunciator 901-3, A-14, TORUS HIGH/LOW LEVEL alarms.

- The 1B RHR room has 8 inches of standing water and rising
- There is a large leak from the 1B RHR suction line from the Torus

After several minutes, with Equipment Operators not yet able to isolate the source of the leak, the following plant conditions exist:

- Reactor Power is 100% and stable.
- Reactor Level is +30 inches and stable.
- Drywell Pressure is 1.15 psig and stable.
- Torus Level is -5 inches and lowering.
- B RHR room water level is approaching the Max Safe Water Level

Per QGA 300, "SECONDARY CONTAINMENT CONTROL," a manual SCRAM is ______(1)____ because _______(2)_______. (Refer to the next page).

A. (1) required
(2) emergency depressurization is anticipated

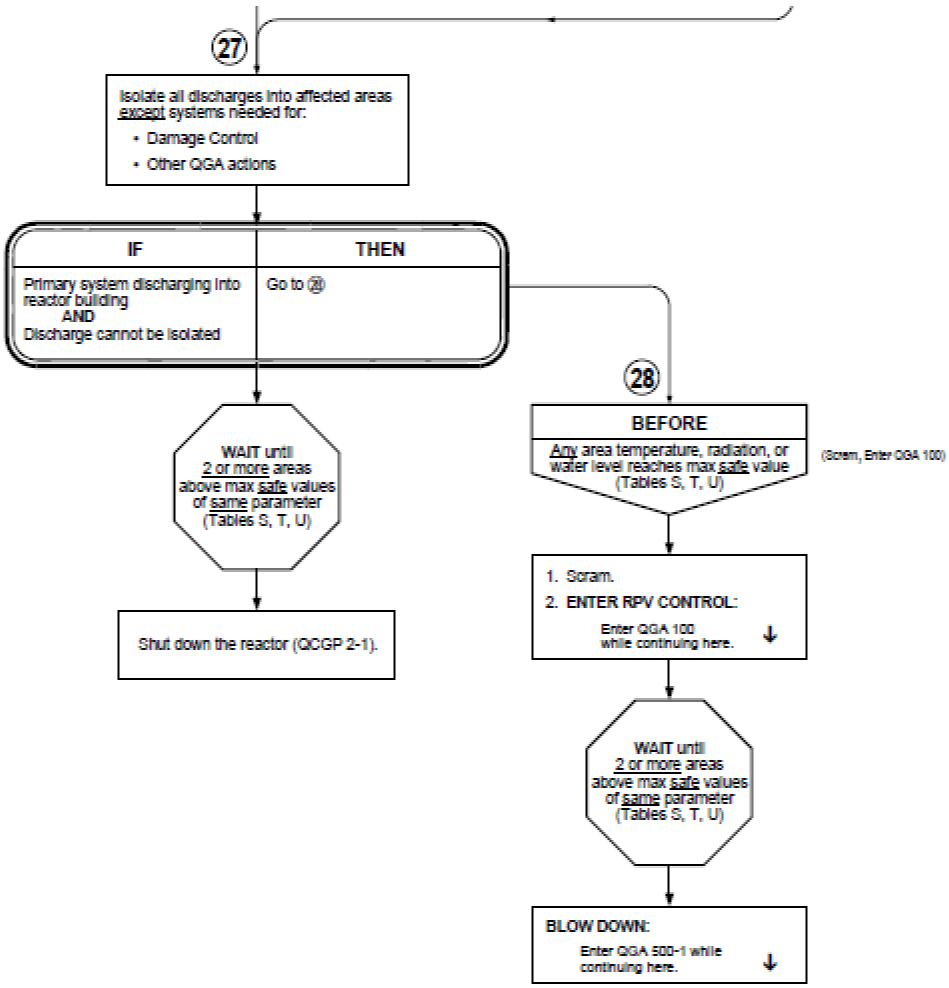
B. (1) NOT required
(2) a primary system is not discharging into the Reactor Building

C. (1) required
(2) secondary containment integrity is being threatened

D. (1) NOT required
(2) an adequate number of ECCS systems remain available

(Question 22)

Excerpt from QGA 300 SECONDARY CONTAINMENT CONTROL



Answer: B

Explanation:

A) Incorrect. Plausible because QGA 300 requires a manual SCRAM in anticipation of an emergency depressurization if the B RHR room exceeded Max Safe Water Level due to an unisolated primary system leak. The Torus is not considered a primary (reactor) system.
B) Correct. QGA 300 would require a normal reactor shutdown IAW QGCP 2-1 if 2 or more areas achieved Max Safe Water Level and would only require a SCRAM on a single area achieving Max Safe Water Level if an unisolated primary system leak were to be occurring. The Torus is not a primary system.
C) Incorrect. Plausible because QGA 300 requires a manual SCRAM if the B RHR room exceeded Max Safe Water Level due to an unisolated primary system leak. Secondary containment integrity remains intact during this event.
D) Incorrect. Plausible because QGA 300 requires a manual SCRAM if the B RHR room exceeded Max Safe Water Level due to an unisolated primary system leak, and even if 1B RHR were inoperable an adequate number of ECCS remain available.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295036EK3.02	
Importance Rating	2.8	

K/A Statement: SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Reactor SCRAM (CFR 41.5)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
1. QGA 300, Secondary Containment Control, Rev. 13
2. QCAN 901-4, D-18, RX BLDG FLOOR DRAIN SUMP B HIGH LEVEL

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0001-K29

Question Source:
Bank # X
Modified Bank # ______
New ______

Question History: Last NRC Exam Fitzpatrick NRC ILT Exam 2012

Question Cognitive Level:
Memory or Fundamental Knowledge ___
Comprehension or Analysis X

Level of difficulty: (1-5) 3

10 CFR Part 55 Content:
55.41 5
55.43 _____

Comments: Question revised to reflect QC specifics.

Question: 28

Initial Conditions:

- Unit 1 was operating at 100% power.
- Core Spray Pump 'B' is out of service for motor replacement.

An unisolable Reactor Coolant System (RCS) leak occurs concurrently with a Loss of Offsite Power (LOOP).

- The 1/2 Emergency Diesel Generator (EDG) failed to start.
- All available ECCS started due to HIGH Drywell Pressure.
- RPV Water Level is being maintained 0-48 inches with HPCI.
- HPCI is also being used to conduct a controlled cooldown of the RPV.
- The HPCI High Torus Level Suction Transfer has been defeated.
- RHR is aligned for Torus Cooling and Torus Spray operation.

Subsequently HPCI trips and isolates. The RPV cooldown is continued using the SRVs.

Which one of the following describes the expected trend in Torus water level following the isolation/trip of the HPCI system?

Torus water level will…

A. Rise continuously until the RPV is completely depressurized.

B. Rise until RPV pressure decreases below the discharge pressure of the running ECCS pumps, then lower as long as the ECCS injection rate is greater than the RCS leak rate

C. Lower continuously until all remaining ECCS pumps are shutdown.

D. Lower until the ECCS injection rate is greater than the RCS leak rate, then rise until the RPV is completely depressurized.

Answer: B

Explanation:

A) Incorrect: After HPCI trips, Torus level will continue to rise due to the leakage from the RCS leak and SRV discharge, but only until RPV pressure drops below the discharge pressure of the running RHR pumps.
B) Correct: Torus level will continue to rise due to the leakage from the RCS leak and SRV discharge, but only until RPV pressure drops below the discharge pressure of the running ECCS pumps. When the ECCS injection flow rate is greater than the RCS leak rate, Torus water level will begin to lower and continue to lower until an equilibrium is established between ECCS flow rate and the RCS leak rate.
C) Incorrect: Torus level will continue to rise due to the leakage from the RCS leak and SRV discharge until the ECCS discharge pressure is greater than RPV pressure.
D) Incorrect: Torus level will continue to rise due to the leakage from the RCS leak and SRV discharge until the ECCS discharge pressure is greater than RPV pressure and will lower only as long as the ECCS injection rate is greater than the RCS leak rate.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	203000A1.05	
Importance Rating	3.8	

K/A Statement: RHR/LPCI: Injection Mode: Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including: Suppression pool level (CFR: 41.5 / 45.5)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. LN-1000 Residual Heat Removal System training plan, Rev. 20
2. QGA-200 Primary Containment Control, Rev. 11
3. L-QGA 200, Primary Containment Control, Rev. 011
4. USAR Chapter 6

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-1000-K02a DESCRIBE the major flowpaths/valve lineups for the following RHR modes of operation:b. LPCI (Injection)

Question Source:

Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	X
Level of Difficulty:	(1-5)	4
10 CFR Part 55 Content:	55.41	5
	55.43	______

Comments:

Question: 29

Unit 1 is in Mode 3 cooling down to enter a refueling outage.

- Reactor coolant temperature is 300ºF.
- ‘A’ RHR loop is operating in Shutdown Cooling Mode, aligned to the ‘A’ Reactor Recirculation loop.
- Both Reactor Recirculation pumps are running.
- MOV 1-3702 RBCCW Drywell Supply valve fails closed and cannot be reopened.

After the operator trips the Reactor Recirculation Pumps,

(1) The operator is required to ________________.

(2) The reason for these actions is to _______________.

A. 1) align/start the ‘B’ RHR loop in Shutdown Cooling mode to the ‘B’ Reactor Recirculation loop
   2) balance the reactor core cooling effect

B. 1) raise reactor coolant level to > 90 inches but < 100 inches, and measure reactor vessel metal temperature once per hour
   2) prevent reactor coolant system temperature stratification

C. 1) raise the reactor coolant level to the reactor vessel flange
   2) prevent reactor coolant system temperature stratification

D. 1) isolate the A RHR loop and re-establish the main condenser if necessary; maintain pressure and temperature control with the bypass valves
   2) balance the reactor core cooling effect

Answer: B

Explanation:

A) Incorrect. Plausible as aligning the other RHR loop for SDC will remove decay heat, but it will not address temperature stratification.
B) Correct. Raising level to between 90-100 inches and taking hourly vessel metal temperatures per QCOP 1000-17 prevents reactor coolant system temperature stratification.
C) Incorrect. Plausible as raising level will address temperature stratification, but raising level to the vessel flange (>100 inches) will flood the steam lines which is undesirable.
D) Incorrect. Isolating the A RHR loop is not required under these conditions and steaming the reactor to the main condenser is not the appropriate actions under present plant conditions. Plausible because steaming the reactor to the condenser is a required action if SDC is lost and cannot be promptly restored; however SDC has not be lost.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	205000 K6.05	
Importance Rating	3.2	

K/A Statement: SHUTDON COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): Knowledge of the effect that a loss of the following will have on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): Component cooling water systems (CFR 41.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QCOA 3700-03, RBCCW High Temperature, Rev. 09
2. QCOP 1000-05, Shutdown Cooling Operation, Rev. 54
3. QCOP 1000-17, Shutdown Cooling, Reactor Temperature Trending, Rev. 15

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-3700-K24

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge ___
Comprehension or Analysis X

Level of difficulty: (1-5) 3

10 CFR Part 55 Content:
55.41 7
55.43 ______

Comments:

Question: 30

Unit 1 was at rated power when a slowly rising high Drywell Pressure condition caused a SCRAM and HPCI initiation.

- HPCI subsequently tripped on high reactor water level
- Reactor water level is 20 inches and slowly lowering.

- A loss of HPCI room cooling resulted in an automatic isolation of the HPCI system.
- HPCI room cooling is subsequently restored and room temperatures have returned to normal.

- Drywell pressure remains above 3 psig.

Which of the following describes the minimum operator action(s) required to reestablish HPCI injection?

A. NO additional operator action is required.

B. Reset AC and DC TRIP LOGIC.

C. Reset the AC and DC TRIP LOGIC; AND
manually reopen the HPCI system isolation valves; AND
manually restart the HPCI Auxiliary Oil Pump.

D. Reset the AC and DC TRIP LOGIC; AND
manually reopen the HPCI system isolation valves; AND
manually restart the HPCI Auxiliary Oil Pump; AND
Reset the HPCI TURB TRIP.

Answer: D

Explanation:

A) Incorrect – Plausible since no operator action would be required if the system had not isolated. The isolation logic must be reset, the isolation valves must be manually reopened, the AOP must be restarted, and the turbine trip reset pushbutton must be depressed.
B) Incorrect – The isolation valves must be manually reopened, the AOP must be restarted, and the turbine trip reset pushbutton must be depressed.
C) Incorrect – The turbine trip reset pushbutton must also be depressed.
D) Correct – Once the listed actions are taken, and with an initiation signal present, the system will restart and inject when RPV water level drops to -59 inches.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	206000A3.09	
Importance Rating	4.2	

K/A Statement: High Pressure Coolant Injection System: Ability to monitor automatic operations of the HIGH PRESSURE COOLANT INJECTION SYSTEM including: Response to system isolation (CFR: 41.7 / 45.7)

Technical Reference(s): LN-2300 HPCI, Rev. 21
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-2300-K07 LIST the signals which cause a HPCI System auto initiation including setpoints. DESCRIBE how they are reset.

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ______

Level of Difficulty: (1-5) 3

10 CFR Part 55 Content:
55.41 ______
55.43 ______

Comments:

Question: 31

Unit 2 is at 75% RTP when a Loss of Offsite Power occurs. Both the ½ EDG and Unit 2 EDG start and power busses 23-1/28 and 24-1/29 respectively. When the busses are reenergized, the MCC 28-1A feed breaker fails open. Which component will lose electrical power as a result?

A. Core Spray inboard injection valve MOV 2-1402-25A

B. ½ Diesel Generator Cooling Water Pump

C. RHR inboard injection valve MOV 2-1001-29A

D. 125 VDC Battery Charger 2A

Answer: A

Explanation:

A) Correct. The power supply for the A train Core Spray injection valves is MCC 28-1A.
B) Incorrect. Plausible as the power supply for the ½ DGCWP is Bus 28.
C) Incorrect. Plausible as the power supply for the A train RHR injection valves is MCC 28-5/29-5.
D) Incorrect. Plausible as the power supply for the 2A 125 VDC Battery Charger is MCC 28-2.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	209001 K2.02	
Importance Rating	2.5	

K/A Statement: LOW PRESSURE CORE SPRAY SYSTEM: Knowledge of electrical power supplies to the following: Valve power (CFR 41.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	1. LIC 1400, Core Spray 2. LN 6500, 4KV/480V Distribution

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-1400-K19

Question Source:
Bank # ____
Modified Bank # ____
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ____

Level of difficulty: (1-5) 2

10 CFR Part 55 Content:
55.41 7
55.43 ____

Comments:

Question: 32

Unit 2 was at 100% power.

On Panel 902-5, the SBLC Initiation switch is placed in the SYS 1 position.

With the above conditions,

MO 2-1201-2, RWCU Suction DW Inboard Isolation valve, is expected to be _____.
MO 2-1201-5, RWCU Suction DW Outboard Isolation valve, is expected to be _____.
MO 2-1201-80, RWCU Return Isolation Valve, is expected to be _____.

A. OPEN
CLOSED
CLOSED

B. CLOSED
OPEN
OPEN

C. CLOSED
OPEN
CLOSED

D. CLOSED
CLOSED
CIOSED

Answer: D

Explanation:

A) Incorrect. Plausible if examinee believes that only valves 5 and 80 close with actuation of SYS 1 and that valve 2 closes with SYS 2.
B) Incorrect. Plausible if examinee believes that only valve 2 closes with actuation of SYS 1 and valves 5 and 80 close with SYS 2
C) Incorrect. Plausible if examinee believes that valve 2 and 80 close with actuation of SYS 1 and the valve 5 closes with SYS 2.
D) Correct. At Quad Cities, both inboard and outboard RWCU suction and return isolation valves receive isolation signals on SBLC initiation.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	211000K1.05	
Importance Rating	3.4	

K/A Statement: Standby Liquid Control System: Knowledge of the physical connections and/or cause effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: RWCU (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. Lesson Plan LN-1200 RWCU, Rev 8
2. FSAR Section 9.3.5 SBLC System Rev.10
3. Lesson Plan LN-1100 SBLC, Rev. 19

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-1200-K11; List the signals which will cause RWCU system isolations.

Question Source:
Bank # ___
Modified Bank # X (Note changes or attach parent)
New ___

Question History: Last NRC Exam Hatch 2015 #9

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ___

Level of Difficulty: (1-5) 2

10 CFR Part 55 Content:
55.41 3, 6, 7, 8
55.43 ___

Comments:
Changed the Plant Hatch valve and panel numbering to Quad Cities numbering.
Quad Cities valves operate differently than Plant Hatch valves (both close).

Question: 33

Unit 2 was operating at 100% power when a loss of Bus 28 occurred. The Bus 28 trip was determined to be inadvertent and Bus 28 was restored five minutes later.

During restart of the 2A RPS MG set, the Equipment Operator is not able to establish 120 VAC at the output of the 2A RPS MG set.

Which of the following condition could be preventing the 2A RPS MG Set from generating its normal output voltage?

A. EPA 2A-1 did not close due to the undervoltage condition

B. RPS Reserve feed interlock operation following Bus 28 outage

C. Control Room Operators did not reset the RPS Channel A ½ SCRAM

D. Operator did not depress the AUXILIARY RESET pushbutton following restart of the 2A RPS MG Set.

Answer: D

Explanation:

A) Incorrect. Plausible as not closing EPA 2A-1 would prevent voltage from being sensed on the 'A' RPS bus from the 2A RPS MG set. 2A RPS MG set is upstream of EPA 2A-1 and its output voltage is therefore not affected by it.
B) Incorrect. Plausible as the Reserve Feed mechanical interlock would prevent feeding the 'A' RPS bus from the 2A RPS MG set if the reserve feed had been aligned to the 'A' RPS bus. Interlock does not affect RPS MG set operation, though.
C) Incorrect. Plausible as the loss of Bus 28 and therefore the 2A RPS MG set caused a ½ SCRAM to occur as RPS Bus 'A' deenergized. The ½ SCRAM cannot be reset until the RPS 'A' bus is reenergized from the either the 2A RPS MG set or the Reserve feed.
D) Correct. During the 2A RPS MG set restoration process, the Operator must depress the AUXILIARY RESET pushbutton to bypass the undervoltage MG set trip and therefore enable the MG set to generate 120 VAC.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	____
	Group #	1	____
	K/A #	212000 A1.01	
	Importance Rating	2.8	

K/A Statement: REACTOR PROTECTION SYSTEM: Ability to predict and/or monitor changes in parameters associated with operating the REACTOR PROTECTION SYSTEM controls including: RPS motor-generator output voltage (CFR 41.5)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QCOP 7000-04, Unit 2 Reactor Protection System MG Sets, Rev. 2

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0500-K21d(4)

Question Source:
- Bank # ____
- Modified Bank # ____
- New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
- Memory or Fundamental Knowledge X
- Comprehension or Analysis ____

Level of difficulty: (1-5) 3

10 CFR Part 55 Content:
- 55.41 5
- 55.43 ____

Comments:

Question: 34

Unit 1 is in startup with the MODE switch in STARTUP and Control Rod pulls in progress.

All IRMs are on range 2, with the following indications:

CHANNEL	READING
IRM 11	60
IRM 12	55
IRM 13	45
IRM 14	70
IRM 15	60
IRM 16	45
IRM 17	50
IRM 18	50

When the IRM 14 range switch is turned one position to the right, IRM 14 displays a reading of 1.

What is(are) the result(s) AND appropriate action(s) to take?

A. IRM DOWNSCALE alarm ONLY.
Continue with the startup.

B. IRM DOWNSCALE alarm and ROD BLOCK.
Bypass IRM 14 and continue with the startup.

C. IRM DOWNSCALE alarm and ½ scram.
Bypass IRM 14 and continue with the startup.

D. IRM DOWNSCALE alarm and ½ scram.
Stop control rod movement. Document the condition. Direct I&C to begin troubleshooting. Do not resume rod motion until the abnormal indications subside.

Answer: B

Explanation:

A) Incorrect. Plausible if the examinee does not recall that the DOWNSCALE Rod Block is not bypassed on Range 2 and above. Additionally, it is not likely that the startup would continue without bypassing IRM 14, since the instrument failed downscale (indication should have been approximately 20 (70/3.16).
B) Correct. Since the range switch is no longer on Range 1, the Rod Block will occur and rod motion cannot continue until IRM 14 is bypassed.
C) Incorrect. Plausible because a CHANNEL A IRM UPSCALE OR INOP alarm will also result in a ½ scram.
D) Incorrect. Plausible since the follow on actions are those for IRM spiking which was not indicated in this scenario. An IRM downscale will not cause a ½ scram but an upscale will.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	215003A2.06	
Importance Rating	3.0	

K/A Statement: Intermediate Range Monitor (IRM) System: Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Faulty range switch (CFR: 41.5 / 45.6)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QCAN 901(2)-5 C-5, Rev. 5
2. QCOA 0700-06, Rev. 2

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0702-K21: Given various plant conditions, PREDICT how Intermediate Range Monitor System/plant parameters will respond to manipulation of the following Intermediate Range Monitor System controls: a.) Range Switch

Question Source:
Bank # X
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam Dresden 2009 #48

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	__X__
Level of Difficulty:	(1-5) __3__	
10 CFR Part 55 Content:	55.41 _5___	
	55.43 _____	

Comments: Changed original question to match Quad Cities terminology and procedures.

Question: 35

A reactor start-up is in progress on Unit 1. The reactor is critical with a 300 second period.

While selecting a control rod for withdrawal, annunciator 901-5, C-3, "ROD OUT BLOCK" is received.

The following indications on the 901-5 panel are displayed:

	SRM 21	SRM 22	SRM 23	SRM 24
CPS	300	320	280	320
RETRACT PERMIT	LIT	LIT	UNLIT	LIT

	IRM 11	IRM 12	IRM 13	IRM 14	IRM 15	IRM 16	IRM 17	IRM 18
UNITS	60	25	35	50	20	65	30	25
RANGE	1	2	2	1	2	1	2	2

Which best describes the plant conditions that will have to be met to bypass this rod block interlock?

A. IRMs 11, 12, 13, 14 ≥ Range 3

B. IRMs 11, 12, 13, 14 ≥ Range 2

C. IRMs 15, 16, 17, 18 ≥ Range 3

D. IRMs 15, 16, 17, 18 ≥ Range 2

Answer: C

Explanation:

A) Incorrect. IRMs 11, 12, 13, 14 are associated with bypassing the Retract Permit rod block for SRMs 21 and 22 only.
B) Incorrect. IRMs 11, 12, 13, 14 are associated with bypassing the Retract Permit rod block for SRMs 21 and 22 only.
C) Correct. IRMs 15, 16, 17, 18 on Range 3 or greater will bypass a Retract Permit rod block from SRM 23.
D) Incorrect. IRMs 15, 16, 17, 18 on Range 3 or greater will bypass a Retract Permit rod block from SRM 23.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	215004 K4.06	
Importance Rating	3.2	

K/A Statement: SOURCE RANGE MONITOR (SRM) SYSTEM: Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: IRM/SRM interlock (CFR 41.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. TRM Table T3.3.a-1
2. QCAN 901-5, C-3, Rod Out Block, Rev. 11

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0701-K19

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge ___
Comprehension or Analysis X

Level of difficulty: (1-5) 3

10 CFR Part 55 Content:
55.41 7
55.43 ______

Comments:

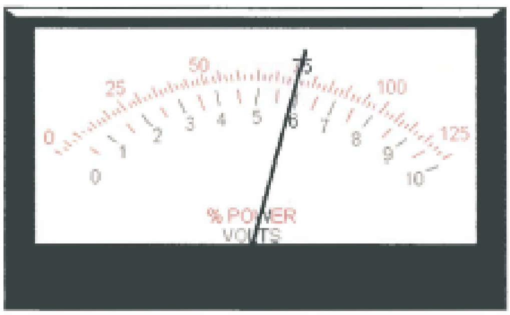
Question: 36

Unit 1 was operating at 100% reactor power when the following Annunciator ALARMS:

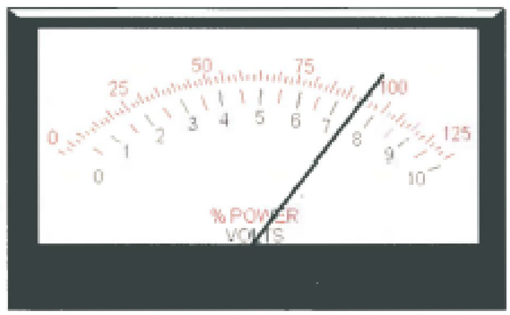
- 901-5 D-6, NEUTRON MON FLOW UNIT OFF NORMAL

A review of the Recirc Flow inputs to the following APRMs is shown below:

NOTE: APRM METER FUNCTION SWITCH IS SELECTED TO THE ‘FLOW’ POSITION



APRM CH 1



APRM CH 4

The RPS scram setpoint for APRM Flow Biased Neutron Flux-High is set at 0.56 W_D + 66.0%.

NO operator actions have been taken.

Which one of the following conditions is expected for the given indications?

A. Half scram on RPS A ONLY

B. Rod out block ONLY

C. Half scram on RPS A and a rod out block

D. APRM flow biased scram setpoint on RPS B HIGHER than normal

Answer: B

Explanation:

The Flow Converter Reference Off Normal Rod Block setpoint is 10% or greater mismatch between channels.

The graphics show APRM 1 with a recirc flow signal at 75% and APRM 4 at 95%, producing a mismatch of 20%.

The rod block being generated is from the Reactor Manual Control System, and will prevent any rod from being withdrawn only.

The RPS scram setpoint for APRM High-High (flow biased) is 0.56WD + 66.0%, where 'WD' is equal to the flow signal.

With the failed APRM 1 Flow signal at 75%, the RPS A scram setpoint would be 108.0%. Since actual reactor power is at 100% as stated in the stem, a scram setpoint would still not be exceeded.

A) Incorrect: Plausible if the lowered flow signal resulted in the scram setpoint being below the indicated power level, and if the 20% flow mismatch between RPS channels (APRM 1 - RPS A / APRM 4 - RPS B) is not recognized.

C) Incorrect: Plausible if the lowered flow signal resulted in the scram setpoint being below the indicated power level.

D) Incorrect: Plausible because this would be the correct answer if assumed that APRM 4 was failed.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	215005K3.03	
Importance Rating	3.3	

K/A Statement: Average Power Range Monitor/Local Power Range Monitor System: Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following: Reactor manual control system (CFR: 41.7 / 45.4)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
QCAN 901-5 D-6, Rev. 6
QCAN 901-5 B-11, Rev. 10

Proposed references to be provided to applicants during examination: None

Learning Objective:	SR-0703-K22 Given a LPRM/APRM System operating mode and various plant conditions, PREDICT how the LPRM/APRM System and plant parameters will be impacted by APRM output fails high/low.		
Question Source:	Bank #	X	
	Modified Bank #	____	(Note changes or attach parent)
	New	____	
Question History:	Last NRC Exam	Quad Cities 2014 #37	
Question Cognitive Level:	Memory or Fundamental Knowledge		____
	Comprehension or Analysis		X
Level of Difficulty:	(1-5)	3	
10 CFR Part 55 Content:	55.41	7	
	55.43	____	

Comments:

Question: 37

Unit 1 is operating at full power, approximately one week after a refueling outage. Given these conditions, which LPRM detector will read the lowest power output?

A. A

B. B

C. C

D. D

Answer: D

Explanation:

A) Incorrect. Void fraction is very low in the bottom of the core and axial flux will be higher low in the core. Therefore, detectors lower in the core will read higher.
B) Incorrect. Void fraction will be relatively low and axial flux will be relatively high.
C) Incorrect. Void fraction will be higher but not as high as the void fraction near detect D in the top of the core. Detector C located higher in the core, but lower than detector D.
D) Correct. The void fraction is greatest at the top of the core and the axial flux will be the lowest.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	215005 K5.02	
Importance Rating	2.7	

K/A Statement: AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM: Knowledge of the operational implications of the following concepts as they apply to AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM: Effects of voids on LPRM indication (CFR 41.5)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QCOP 0700-03, Local Power Range Monitoring (LPRM) Operation, Rev. 20
2. LIC-RXTH, Reactor Theory; Rev 8

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0703-K20

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge ___
Comprehension or Analysis X

Level of difficulty: (1-5) 2

10 CFR Part 55 Content:
55.41 5
55.43 ______

Comments:

Question: 38

The Unit 1 Reactor Core Isolation Cooling (RCIC) System is in the pressure control mode of operation in accordance with QCOP 1300-02 RCIC System Manual Startup (Injection/Pressure Control), following a reactor scram thirty minutes ago.

- RCIC FIC-1-1340-1, RCIC flow controller is in Automatic.
- RCIC FIC-1-1340-1, RCIC flow controller is set to 400 GPM.
- Reactor Pressure is stable.

Shortly after the alignment was established the following occurs:

- RCIC FT-1-1360-4, discharge flow transmitter has failed low such that the flow sensed by RCIC FIC-1-1340-1 is 0 GPM irrespective of actual RCIC flow.

How do Reactor Pressure and RCIC speed respond to this failure?

	Reactor Pressure	RCIC Speed
A.	Slowly rises	Rises and stabilizes at 4500 rpm
B.	Slowly rises	Lowers to idle speed
C.	Slowly lowers	Rises and stabilizes at 4500 rpm
D.	Slowly lowers	Lowers to idle speed

Answer: C

Explanation:

A) Incorrect: Pressure rising would be true if the RCIC turbine slowed down drawing less steam. The speed rising and stabilizing at 4500 rpm is true because the controller will speed the turbine up trying to increase flow. 4500 rpm is the upper limit of the automatic speed control range. The RCIC mechanical overspeed is at 5600 rpm.
B) Incorrect: Pressure rising would be true if the RCIC turbine slowed down drawing less steam. The speed slowing down to idle speed would only be true if the flow controller failed to 500 gpm or the maximum output.
C) Correct. Reactor pressure would lower since the flow sensor failed to 0 GPM which would cause the RCIC turbine to draw more steam as it sped up to 4500 rpm. The speed rising and stabilizing at 4500 rpm is true because the controller will speed the turbine up trying to increase flow.
D) Incorrect. Reactor pressure would lower since the flow sensor failed to 0 GPM which would cause the RCIC turbine to draw more steam as it sped up to 4500 rpm. The speed slowing down to idle speed would only be true if the flow controller failed to 500 gpm or the maximum output.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	217000K3.02	
Importance Rating	3.6	

K/A Statement: Reactor Core Isolation Cooling System (RCIC): Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on following: Reactor vessel pressure (CFR: 41.7 / 45.4)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
LN-1300 RCIC, Rev. 16
QCOP 1300-02 RCIC System Manual Startup (Injection/Pressure Control), Rev. 32

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source:
Bank # ______
Modified Bank # X (Note changes or attach parent)
New ______

Question History: Last NRC Exam 2011 Cooper exam #6

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	x
Level of Difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	7
	55.43	______

Comments: Changed failure so that flow input fails low instead of high. Revised parameters to QC specific values.

Question: 39

With the Unit 2 RCIC system running for a routine surveillance, operators receive annunciator 902-4, G-15, RCIC TRIP THROTTLE VALVE CLOSED and note that the CLOSED light is LIT for the 2-1303B Trip Throttle Valve on the 902-4 vertical panel. No other alarms are received.

Which condition could have caused the indications described above?

A. Reactor water level +48 inches

B. RCIC turbine exhaust pressure 35 psig

C. RCIC turbine speed 5600 rpm

D. RCIC room temperature 180ºF

Answer: C

Explanation:

A) Incorrect. Plausible as reactor water level of +48 inches will close the 2-1301-61 Steam to Turbine Valve and will be indicated by annunciator 902-6, F-11, RFP/TURBINE RX VESSEL HI LEVEL.
B) Incorrect. Plausible as RCIC turbine exhaust pressure of > 30 psig will close the 2-1301-61 Steam to Turbine Valve and will be indicated by annunciator 902-4, D-15, RCIC TURBINE TRIP.
C) Correct. RCIC turbine overspeed ≥ 5600 rpm or manually tripping will result in closing the 2-1303 and will result in the alarm and light indications listed above on the 902-4 panel.
D) Incorrect. Plausible as RCIC room temperature ≥ 170°F will result in a system isolation, but will be indicated by annunciators 902-4, D-15, RCIC TURBINE TRIP and 902-4, B-15, RCIC STEAMLINE ISOLATION.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	217000 A3.06	
Importance Rating	3.5	

K/A Statement: REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): Ability to monitor automatic operations of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) including: Lights and alarms (CFR 41.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	1. QCAN 901-04, G-15, RCIC Trip Throttle Valve Closed, Rev. 03 2. QCAN 901-4, D-15, RCIC Turbine Trip, Rev. 07

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-1300-K06

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ____

Level of difficulty: (1-5) 2

10 CFR Part 55 Content:
55.41 7
55.43 ____

Comments:

Question: 40

Events have occurred on Unit 1, resulting in the following conditions:

- Loss of all offsite power
- HPCI is unavailable
- ALL ERV and SRV position indicating lights are OFF on the 901-3 panel.
- ADS BLOWDOWN INHIBIT SWITCH is positioned to **NORMAL**.
- RPV pressure is 500 psig and lowering at 10 psig/minute.
- Drywell pressure is 1.8 psig and rising 0.1 psig/minute.
- RPV level is -30 inches and lowering 10 inches/minute.

When will Low Pressure ECCS pumps **FIRST** inject?

A. In 6 minutes

B. In 9 minutes

C. In 12 minutes.

D. In 18 minutes.

Answer: D

Explanation:

A.) Incorrect. There is no power to the solenoids for the ERVs or SRV, so they cannot be opened either automatically or manually. The time corresponds to RPV level dropping to -59 inches and a misconception that the 110 second timer will start (the 110 second timer does not start until both a low-low reactor water level and a high drywell pressure exist), with additional time for pressure to go below the shutoff head of the ECCS pumps.
B.) Incorrect. There is no power to the solenoids for the ERVs or SRV, so they cannot be opened either automatically or manually. .The time corresponds to reactor water level reaching a low-low condition (-59") coupled with a high drywell pressure (+2.5 psig in 7 minutes) with the 110 second timer expiring.
C.) Incorrect. There is no power to the solenoids for the ERVs or SRV, so they cannot be opened either automatically or manually. The time corresponds to reactor water level reaching a low-low condition (-59" in 3 minutes) and the expiration of the 8.5 minute timer.
D.) Correct. This corresponds to the amount of time it takes for pressure to drop to the shutoff head of the low pressure ECCS pumps since ADS will not actuate. It takes 18 minutes for pressure to drop less than 325 psig.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	218000K3.01	
Importance Rating	4.4	

K/A Statement: Automatic Depressurization System: Knowledge of the effect that a loss or malfunction of the AUTOMATIC DEPRESSURIZATION SYSTEM will have on following: Restoration of reactor water level after a break that does not depressurize the reactor when required (CFR: 41.7 / 45.4)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. LIC-0203 ADS, Rev. 19
2. QCAN 901-3 G-13, Rev. 2

Proposed references to be provided to applicants during examination: None

Learning Objective: ________________________ (as available)

Question Source:
Bank # X
Modified Bank # ______ (Note changes or attach parent)
New ________

Question History: Last NRC Exam Dresden 2017 #47

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	__X___
Level of Difficulty:	(1-5)	__3__
10 CFR Part 55 Content:	55.41	__7__
	55.43	______

Comments: Altered the answers to correspond with Quad Cities specific information. Deleted reference to ADS DC power failure alarm as the alarm has multiple inputs and would not provide any useful information. Added additional initial conditions which do not alter the basis behind the question but adds some credibility to the postulated event.

Question: 41

Unit 2 is in Mode 3 cooling down to Mode 4 for a forced outage to repair the HPCI system (unavailable).

A transient occurs resulting in the following plant conditions:

• Drywell Pressure	2.0 psig and steady
• Drywell Average Air Temp.	200ºF and slowly rising
• Torus Temperature	100 ºF and slowly rising
• Reactor Pressure	550 psig and slowly lowering
• Reactor Water Level	-60 inches and lowering
• RCIC System	Injecting following auto-start

What actions should operators take regarding use of the Automatic Depressurization System (ADS) and its initiation logic?

A. Inhibit ADS to avoid initiation following time out of the 110 second timer

B. Inhibit ADS to avoid initiation following time out of the 8.5 minute timer

C. Press the ADS TIMER RESET pushbutton to reset the 8.5 minute timer

D. Manually open the ADS valves to depressurize the RPV below 325 psig

Answer: B

Explanation:

A) Incorrect. Plausible as the 8.5 minute timer has started and would be inhibited with the INHIBIT key lock switch, but the 110 second timer has not started since DW pressure is < 2.5 psig.

B) Correct. With a LOCA from a reactor at lower pressure, DW pressure has not reached 2.5 psig. Therefore with a low low reactor water level (<-59 inches) and no high drywell pressure signal (>2.5 psig) the ADS 8.5 minute timer has started but not the 110 second timer.

C) Incorrect. Plausible as pressing the TIMER RESET pushbutton resets the 110 second timer, but does not reset the 8.5 minute timer.

D) Incorrect. Plausible as lowering RPV pressure will allow LPCI and CS to start and inject, but with RPV level well above TAF there is not a need to subject the RPV to the stresses that ADS initiation results in as well as to put additional energy into the containment.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	218000 2.4.9	
Importance Rating	3.8	

K/A Statement: AUTOMATIC DEPRESSURIZATION: Knowledge of low power/shutdown implications in accident (e.g. loss of coolant accident or loss of residual heat removal) mitigation strategies (CFR 41.10)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QGA 100, RPV Control, Rev. 11
2. QGA 200, Primary Containment Control, Rev. 11

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0203-K16 and SR-0203-K21

Question Source:
- Bank # ______
- Modified Bank # ______
- New X

Question History: Last NRC Exam Quad Cities 2011 Cert Exam

Question Cognitive Level:
- Memory or Fundamental Knowledge ___
- Comprehension or Analysis X

Level of difficulty: (1-5) 3

10 CFR Part 55 Content:
- 55.41 10
- 55.43 _____

Comments:

Question: 42

Main Steam Isolation Valve (MSIV) actuator motive force is supplied from the ___(1)___ and the associated solenoid valves are powered from ___(2)___.

	Inboard	Outboard
A.	(1) Drywell Pneumatic System	Instrument Air System
	(2) RPS A	RPS B
B.	(1) Drywell Pneumatic System	Instrument Air System
	(2) RPS B	RPS A
C.	(1) Instrument Air System	Drywell Pneumatic System
	(2) RPS A	RPS B
D.	(1) Instrument Air System	Drywell Pneumatic System
	(2) RPS B	RPS A

Answer: A

Explanation:

A) Correct: Inboard MSIVs receive air operator power from the Drywell Pneumatic System and AC solenoid power from RPS A. Outboard MSIVs receive air operator power from the Instrument Air System and AC solenoid power from RPS B.
B) Incorrect: While Inboard MSIVs receive air operator power from the Drywell Pneumatic System and Outboard MSIVs receive air operator power from the Instrument Air system, the AC solenoid power is reversed.
C) Incorrect: While Inboard MSIVs receive AC solenoid power from RPS A and Outboard MSIVs receive AC solenoid power from RPS B, the air operator power is reversed.
D) Incorrect: Both the air operator power source and solenoid power sources are reversed for Inboard and Outboard MSIVs.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	223002K4.01	
Importance Rating	3.0	

K/A Statement: Primary Containment Isolation System/Nuclear Steam Supply Shut-Off: Knowledge of PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF design feature(s) and/or interlocks which provide for the following: Redundancy (CFR: 41.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
1. LN-1603 Primary Containment Isolation System, Rev. 14
2. LIC-0250 Main Steam, Rev. 18

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0250-K19 List the plant systems which support the Main Steam System and Describe the nature of support. (Includes power supplies)

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ______

Level of Difficulty: (1-5) 2

10 CFR Part 55 Content:
55.41 7
55.43 ______

Comments:

Question: 43

With Unit 1 operating at full power, a reactor water level control transient occurred causing reactor water level to lower to -5 inches before being recovered to the normal operating level band.

(1) With regards to the Primary Containment Isolation System (PCIS) only, which isolations would have occurred?

AND

(2) On which section of the 901-3 panel can the completion status of those isolations be observed?

A. (1) PCIS Groups I, II, and III
(2) vertical section

B. (1) PCIS Groups II and III ONLY
(2) bench board section

C. (1) PCIS Groups I, II, and III
(2) bench board section

D. (1) PCIS Groups II and III ONLY
(2) vertical section

Answer: D

Explanation:

A) Incorrect. (1) Water level did not drop low enough to actuate PCIS Group I; therefore, only PCIS groups II, III actuate. (2) Isolation status of all PCIS Group II and III Containment Isolation Valves can be verified using the Isolation Valve Position mimic on the vertical section of panel 901-3.
B) Incorrect. (1) PCIS groups II, and III would actuate under these conditions. (2) While several of the isolation valve positions could be verified on the bench board section of panel 901-3, not all of the PCIS Group II and III valve positions are indicated on this section.
C) Incorrect. (1) Water level did not drop low enough to actuate PCIS Group I; therefore, only PCIS groups II, III actuate. (2) While several of the isolation valve positions could be verified on the bench board section of panel 901-3, not all of the PCIS Group II and III valve positions are indicated on this section.
D) Correct. (1) PCIS groups II, and III would actuate under these conditions. (2) Isolation status of all PCIS Group II and III Containment Isolation Valves can be verified using the Isolation Valve Position mimic on the vertical section of panel 901-3.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	223002 A4.06	
Importance Rating	3.6	

K/A Statement: PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF: Ability to manually operate and/or monitor in the control room: Confirm initiation to completion (CFR 41.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QCAP 0200-10, Emergency Operating Procedure (QGA) Execution Standards, Rev. 54
2. QCAN 901-5, A-8, Group 2 ISOL CH Trip, Rev. 14
3. QCAN 901-5, B-8, RWCU PCIS RX Water Level LO CH Trip, Rev. 11

Proposed references to be provided to applicants during examination: None

Learning Objective: SRN-1601-K20

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
Level of difficulty:	(1-5) 2	
10 CFR Part 55 Content:	55.41 7	
	55.43	

Comments:

Question: 44

Unit 1 was operating at 100% power when a Loss of Offsite Power occurred.

- A controlled cooldown is in progress.
- A failure in the Drywell Pneumatics system has resulted in depressurization of the system.

The PREFERRED method to continue the cooldown is to use the ____.

A. Target Rock Relief Valve

B. Electromatic Relief Valves

C. Main Steam Line drains to the main condenser.

D. Reactor Head Vents

Answer: B

Explanation:

A) Incorrect. Manual operation of the Target Rock Relief Valve requires a pneumatic supply and though the supply line has an accumulator, it is sized for only 5 valve cycles. QGA 100 states to minimize its use upon a loss of Drywell Pneumatics.
B) Correct. Procedure QCOP 0203-01 Reactor Pressure Control Using Manual Relief Valve Actuation states that with a loss of drywell pneumatics, to use the Electromatic relief valves 1-203-3B, 3C, 3D, and 3E.
C) Incorrect. The main condenser is unavailable, so opening the main steam line drains would not be preferred. This is plausible since this procedure exists and the applicant may not recall that the main condenser is unavailable during the Loss of Offsite Power. (QCOP-0250-05 Reactor Pressure Control Using Main Steam Line Drains)
D) Incorrect. Opening the Reactor Head Vents to reduce pressure is an emergency procedure only used when directed by the QGAs. (QCOP 0220-02 Emergency Depressurization with Reactor Head Vents).

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	239002K6.02	
Importance Rating	3.4	

K/A Statement: Relief/Safety Valves: Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES: Air (Nitrogen) supply (CFR: 41.7 / 45.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. LN-4702 Drywell Pneumatic System, Rev. 6
2. LIC-0203 ADS, Rev. 19
3. LIC-0250 Main Steam, Rev. 18

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0203-K23 Given various plant conditions, PREDICT how the ADS logic/valves will be impacted by the following support system failures: c. Loss of drywell pneumatics.

Question Source:
Bank # ________
Modified Bank # ________ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	___
	Comprehension or Analysis	X
Level of Difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	7
	55.43	___

Comments:

Question: 45

Initial Conditions:

- Unit 1 is operating at 100% reactor power.
- The Feedwater Level Control System is in 3-Element Control with all Feedwater Regulating Valve (FRV) controllers in AUTOMATIC

An event occurs resulting in the following:

- The ‘B’ reactor feed pump flow indication fails high to 6.0E6 lbm/hr
- Annunciator 901-5, G-8, “FEED WTR PUMP MAX CAPACITY” alarms

Assuming NO operator actions, how will the Digital Feedwater Level Control System (DFWLCS) respond?

A. FRV controllers will transfer to MANUAL control.
Operator action will be required to maintain RPV water level.
B. The DFWLCS will transfer to Single-Element Control.
The FRVs will throttle to maintain RPV water level at the Water Level Setpoint.
C. The ‘B’ Reactor Feed Pump Flow input will be deselected.
The DFWLCS will remain in 3-Element Control.
RPV water level will be returned to the Water Level Setpoint.
D. The DFWLCS will transfer to Runout Flow Control mode.
FRVs will throttle closed to limit the feed pump flow at the runout setpoint
Operator action will be required to prevent a reactor SCRAM.

Answer: D

Explanation:

A) Incorrect: FRV controllers will remain in automatic under these conditions. Plausible since FRV controllers will transfer to MANUAL under certain conditions (e.g., if there are less than two valid level channels)
B) Incorrect: FWLCS remains in 3-Element Control. The DFWLCS has a feature, currently disabled at Quad Cities, that would transfer the control system Single Element Control under these conditions, but would limit the FRV demand to the value at the time of the transfer.
C) Incorrect: Unlike the Level and Steam Flow channels, the Feed Flow Channels do not have a Soft Majority Select (SMS) feature.
D) Correct: FRVs will throttle closed due to measured feed flow being greater than measured steam flow, and RPV water level will begin to drop due to actual feed flow being less than measured steam flow. FRVs will continue to throttle closed until: 1) measured feed flow is less than steam flow, OR 2) RPV Level Error overrides the steam-feed flow error. Measured feed flow (and therefore FRV open position) will be limited to the feed pump high flow runout set point, irrespective of the Level Error magnitude. RPV water level will continue to drop, due to actual feed water flow remaining less that steam flow. resulting in lowering reactor level due to the feed flow < steam flow mismatch and eventual reactor SCRAM

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	259002 K4.03	
Importance Rating	2.8	

K/A Statement: REACTOR WATER LEVEL CONTROL SYSTEM: Knowledge of REACTOR WATER LEVEL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Reactor feed pump runout protection: MDFP (CFR 41.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	1. QCOA 0600-09, Main Feed Pump Runout, Rev. 11 2. QCAN 901-5, G-8, FEED WTR PUMP MAX CAPACITY, Rev. 07 3. QCOA 3200-01, Reactor Feed Pump Auto Trip, Rev. 23

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-3200-K23g and SR-3200-K06

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	___
	Comprehension or Analysis	_X_
Level of difficulty:	(1-5) __3__	
10 CFR Part 55 Content:	55.41 __7__	
	55.43 _____	

Comments:

Question: 46

Unit 2 is operating at 100% Rated Thermal Power.

The Standby Gas Treatment (SBGT) Train ‘A’ Fan is OOS for maintenance

The following sequence of events occurs:

1000	RWCU System break in the Unit 2 Reactor Building
1005	Reactor Bldg. Vent Rad Monitors exceed 10 mr/hr
1010	The Supply breaker for MCC 19-4 trips OPEN

At 1008, the Reactor Building Stack release rate will be __(1)__ than at 1004.

At 1015, the Chimney release rate will be __(2)__ 1008.

A. (1) lower
(2) lower than

B. (1) lower
(2) approximately the same as

C. (1) higher
(2) lower than

D. (1) higher
(2) approximately the same as

Answer: A

Explanation:

1000 – With initiation of the RWCU system leak, the RB Ventilation Radiation Monitor levels begin to rise.

1005 -- The Reactor Building ventilation system isolates when the RB Ventilation and the 'B' Train of SBGT starts and directs all RB exhaust to the Chimney. Reactor Building Stack release rate drops dramatically as flow to the stack is terminated by the ventilation system isolation. The plant Chimney release rate will rise due SBGT system flow exhausting to the Chimney.

1010 – The 'B' Train of SBGT loses power when the supply breaker for MCC 19-4 trips. The Plant Chimney release rate drops due to the loss of SBGT flow.

A) Correct. (1) The reactor building release rate drops because the reactor building ventilation isolates. (2) The Chimney release rate drops because all SBGT flow is lost.
B) Incorrect. (1) The reactor building release rate drops because the reactor building ventilation isolates. (2) Plausible because the applicant may not recognize that power to the B train of SBGT is lost at 1010.
C) Incorrect. (1) Plausible because the applicant may not recognize/recall that the RB ventilation isolates when the RB Ventilation Radiation Monitor exceeds 10 mr/hr. (2) The Chimney release rate drops because all SBGT flow is lost.
D) Incorrect. (1) Plausible because the applicant may not recognize/recall that the RB ventilation isolates when the RB Ventilation Radiation Monitor exceeds 10 mr/hr. (2) Plausible because the applicant may not recognize that power to the B train of SBGT is lost at 1010

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	261000K3.02	
Importance Rating	3.6	

K/A Statement: Standby Gas Treatment System: Knowledge of the effect that a loss or malfunction of the STANDBY GAS TREATMENT SYSTEM will have on following: Off-site release rate (CFR: 41.7 /45.6)

Technical Reference(s):
(Attach if not previously provided, including version/revision number.)

1. LIC 7500, Standby Gas Treatment System, Rev. 20
2. QCOP 1600-13 Post-Accident Venting of the Primary Containment (H.7.b), Rev. 29
3. QCOP 7500-01 Standby Gas Treatment System (SBGTS) Standby Operation and Startup, Rev. 21
4. LIC-1702 Chimney Radiation Monitoring, Rev. 12
5. LIC-1701 Process Radiation Monitoring, Rev. 2

Proposed references to be provided to applicants during examination: None

Learning Objective:	SR-1702-K23 Given a Chimney/Stack Radiation Monitoring System operating mode and various plant conditions, PREDICT how the Chimney/Stack Radiation Monitoring System will be impacted by the following failures: a. Loss of Essential Service b. Loss of Instrument Bus c. Loss of 24/48 VDC power d. Loss of MCC 17-1-1 (27-1-1) e. Loss of 27-1 (SPING) f. Extremely high radiation levels g. Loss of 'bug' source
Question Source:	Bank # X Modified Bank # ___ ___ (Note changes or attach parent) New ________
Question History:	Last NRC Exam Hatch 2012 #29
Question Cognitive Level:	Memory or Fundamental Knowledge ______ Comprehension or Analysis X
Level of Difficulty:	(1-5) 3
10 CFR Part 55 Content:	55.41 7 55.43 _____
Comments:	

Question: 47

Unit 1	Unit 2
Mode 4	Mode 1
‘A’ and ‘B’ RHR pumps in SDC mode	100% reactor power
‘A’ and ‘B’ RHR service water pumps running	Bus 23-1 cross-tied to Bus 13-1
TR 12 OOS for maintenance	Bus 24-1 cross-tied to Bus 14-1

Cross-tie parameters:

	13-1/23-1	14-1/24-1
Voltage	4160 VAC	4140 VAC
Current	625 Amps	125 Amps

What actions are required to address distribution load restrictions?

A. Start the Unit 1 SBO DG1 and parallel on to Bus 13-1 to share load with Bus 23-1 through the existing 13-1/23-1 cross-tie.

B. Open BUS 13-1 AND 23-1 TIE GCB breaker and allow the ½ EDG to auto start and power Bus 13-1.

C. Reduce loading on the 13-1/23-1 cross-tie to < 500 Amps if possible. Cross-tie current may not exceed 600 Amps.

D. Reduce loading on the operating unit (Unit 2) to no more than 2 condensate pumps, 2 circulating water pumps, and 1 service water pump.

Answer: C

Explanation:

A) Incorrect. Plausible as you can operate with the SBO diesels across the cross-tie, but these are not the actions required per plant operating procedures. In addition, cross-tie current is high on division 1, but it is not outside the allowable < 600A restriction.
B) Incorrect. Plausible as the ½ EDG would start on undervoltage as felt on the 13-1 bus, but this would result in a loss of SDC which would have to be manually restarted on the EDG. 525A on the 23-1/13-1 cross-tie is not an overcurrent condition.
C) Correct. Cross-tie currents for 13-1/23-1 and 14-1/24-1 are to be < 600A. If >500A should reduce current/loading or monitor twice/hour per QCOP 6500-08.
D) Incorrect. Plausible as these are the actions if Unit 2 were to lose UAT TR21 while cross-tied to Unit 1 safety-buses per QCOP 6500-08.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	262001 A1.04	
Importance Rating	2.7	

K/A Statement: A.C. ELECTRICAL DISTRIBUTION: Ability to predict and/or monitor changes in parameters associated with operating the A.C. ELECTRICAL DISTRIBUTION controls including: Load currents (CFR 41.5)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
1. QCOP 6500-08, 4KV Bus Cross-Tie Operation, Rev. 30

Proposed references to be provided to applicants during examination: None

Learning Objective: SRN-6500-K04 and SRN-6500-K13

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge ___
Comprehension or Analysis X

Level of difficulty: (1-5) 3

10 CFR Part 55 Content:
55.41 5
55.43 ______

Comments:

Question: 48

Unit 2 is at 100% power when the following annunciators ALARM on the 902-8 panel:

- 902-8 B-8, 120/240V AC ESS SERV BUS LOW VOLTAGE
- 902-8 E-9, ESS SERV BUS ON EMERG SPLY
- 902-8 F-8, ESS SERV UPS TROUBLE

Select the answer choice below that completes the following statements:

(1) Power to the ESS Bus was _______________ during the above event.

(2) When the cause of the above event has been corrected, the source of power to the ESS Bus _______________ back to the UPS (Static Switch) Output.

A. (1) uninterrupted
(2) will AUTOMATICALLY transfer

B. (1) uninterrupted
(2) must be MANUALLY transferred

C. (1) momentarily interrupted
(2) will AUTOMATICALLY transfer

D. (1) momentarily interrupted
(2) must be MANUALLY transferred

Answer: D

Answer Explanation

The alarming annunciators indicate that the output of the UPS has failed, resulting in the ABT shift from the ESS UPS (inverter or regulator) supply to the ESS reserve (MCC 28-2) AC supply.

The ABT is "power-seeking" and will not transfer back to the ESS UPS supply when UPS power is restored.

A) Incorrect: The ABT transfer is break before make, therefore power will be momentarily interrupted. The ESS ABT is power seeking and must me manually transferred upon restoration of normal power supply. Plausible if the examinee does not recognize the UPS failure and concludes that the UPS Static Switch transferred from the Normal (inverter) to the Alternate (regulator) AC source. The UPS Static Switch is normal seeking is a make before break transfer.
B) Incorrect: The ABT transfer is break before make, therefore power will be momentarily interrupted. Plausible if the examinee does not recognize the UPS failure and concludes that the UPS Static Switch transferred from the Normal (inverter) to the Alternate (regulator) AC source.
C) Incorrect: The ABT transfer is break before make, therefore power will be momentarily interrupted. Plausible if the examinee does not recognize the UPS failure and concludes that the UPS Static Switch transferred from the Normal (inverter) to the Alternate (regulator) AC source. The UPS Static Switch is normal seeking and automatically transfers back to the Normal (inverter) source.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262002A2.01	
	Importance Rating	2.8	

K/A Statement: Uninterruptable Power Supply (A.C./D.C.): Ability to (a) predict the impacts of the following on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Under voltage (CFR: 41.5 / 45.6)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
1. LN-6800 Ess. Service/Instrument Bus System, Rev. 17
2. QOP 6800-03, Essential Service System

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-6800-K10 List the signals which cause the Essential Service/Instrument Bus Systems to auto transfer including purpose and setpoints. DESCRIBE how they are bypassed and how they are reset.

Question Source:	Bank #	________	
	Modified Bank #	________	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge	______	
	Comprehension or Analysis	X	
Level of Difficulty:	(1-5)	4	
10 CFR Part 55 Content:	55.41	5	
	55.43	______	

Comments:

Question: 49

Both units were operating at near rated power with the Unit 1 Safety-Related 250 VDC battery undergoing an equalization charge per QCOP 6900-51, “Unit 1 Battery Equalizing Charges.”

During the battery charge, the battery room ventilation system tripped due to an overcurrent condition.

What are the initial implications of the above failure?

A. Hazardous levels of Hydrogen may accumulate, which could lead to an explosive atmosphere in the Unit 1 Battery Room

B. Cell temperatures may increase, causing the battery voltage to increase adversely affecting safety-related loads

C. Low safety-related 250VDC bus voltage and therefore inoperable HPCI and RCIC due to a loss of Unit 1 Battery Charger

D. Oxygen levels may decrease in the Unit 1 Battery Room, making the room unsafe for habitability

Answer: A

Explanation:

A) Correct. Battery Room Ventilation must be operating OR alternate ventilation supplied during an equalizing battery charge. Hazardous levels of hydrogen may accumulate if Battery Room Ventilation is NOT working. Loss of 15-2 results in loss of Unit 1 Battery Room ventilation.
B) Incorrect. As room temperature rises battery voltage lowers. Additionally significant temperature deviation is not expected with loss of ventilation.
C) Incorrect. Plausible as non-safety-related 250VDC battery ‘A’ charger is powered from MCC 15-2.
D) Incorrect. Oxygen levels are not affected by ventilation fan operations.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	263000 K5.01	
Importance Rating	2.6	

K/A Statement: D.C. ELECTRICAL DISTRIBUTION: Knowledge of the operational implications of the following concepts as they apply to D.C. ELECTRICAL DISTRIBUTION: Hydrogen generation during battery charging (CFR 41.5)

Technical Reference(s): (Attach if not previously provided, including version/revision number.
1. QCOP 6900-51, Unit 1 Equalizing Battery Charges, Rev. 51

Proposed references to be provided to applicants during examination: None

Learning Objective: SRN-6900-K15

Question Source:
Bank # X
Modified Bank # ____
New _

Question History: Last NRC Exam Dresden NRC ILT Exam 2016

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ____

Level of difficulty: (1-5) 2

10 CFR Part 55 Content:
55.41 5
55.43 ____

Comments:

Question: 50

The Unit 2 250 VDC battery charger is O.O.S. with the Unit 2 250 VDC battery being supplied from the 1/2 charger powered from Unit 1.

The Unit 1 250 VDC battery charger is in service and being powered from its normal power supply.

MCC 19-2 loses power and cannot be restored.

What are the expected control room indications?

A. Only the 901-8 panel indication for 250 VDC battery voltage will begin lowering.

B. Only the 902-8 panel indication for 250 VDC battery voltage will begin lowering.

C. 901-8 and 902-8 panel indications for 250 VDC battery voltage will begin lowering.

D. 901-8 and 902-8 panel indications for 250 VDC battery voltage will remain constant.

Answer: A

Explanation:

A) Correct. All battery charger power is lost to Unit 1 250 VDC. Therefore, the 250 VDC battery begins discharging and voltage will begin lowering.
B) Incorrect. This is plausible if the applicant forgets that MCC 18-2 is the Unit 1 power supply to Battery charger 1/2. The Unit 2 250 VDC battery voltage would decrease if power were lost to the battery charger.
C) Incorrect. This is plausible if the applicant thinks that MCC 19-2 provided power to both the Unit 1 Battery Charger and the Unit 1/2 Battery Charger.
D) Incorrect. This is plausible if the applicant thinks that neither battery charger is supplied by MCC 19-2.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	263000A4.03	
Importance Rating	2.7	

K/A Statement: D.C. Electrical Distribution: Ability to manually operate and/or monitor in the control room: Battery discharge rate (CFR: 41.7 / 45.5 to 45.8)

Technical Reference(s): LN-6900 DC Distribution and Batteries, Rev. 21
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: SRN-6900-K26 (as available)

Question Source:
Bank # ____
Modified Bank # X (Note changes or attach parent)
New ____

Question History: Last NRC Exam Dresden 2017 #48

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ____

Level of Difficulty: (1-5) 3

10 CFR Part 55 Content:
55.41 7
55.43 ____

Comments:

Changed the battery charger(s) that would lose power.

Question: 51

Unit 1 is operating at 100% power when Turbine Bldg. Res Bus 1B-1 125VDC power trips on overcurrent. Before any operator action occurs, Unit 1 experiences a LOCA causing drywell pressure to exceed 2.5 psig.

What is the status of the Unit 1 EDG?

A. The Unit 1 EDG is running unloaded at 60 Hz and 4160 V

B. The Unit 1 EDG is supplying power to Bus 14-1

C. The Unit 1 EDG is running at 900 rpm with no voltage developed at the generator

D. The Unit 1 EDG is not running

Answer: D

Explanation:

A) Incorrect. Plausible as the ½ EDG has an auto transfer switch which will enable the ½ EDG starting logic and components to be powered by the opposite Unit's 125 VDC. The Unit 1 EDG would normally start and run unloaded under LOCA conditions, but without 125 VDC it will NOT be able to start.

B) Incorrect. Plausible as EDG normally powers Bus 14-1, but it would not automatically power Bus 14-1 without a LOOP signal present, and in this case it cannot start without 125 VDC available to it.

C) Incorrect. Plausible as the field flash is not available without 125 VDC, but EDG would not be able to start (starting logic, starting air solenoid, governor booster pump, fuel priming pump) as well without 125 VDC.

D) Correct. Unit 1 EDG cannot start (starting logic, starting air solenoid, governor booster pump, fuel priming pump, field flash) without Unit 1 125 VDC from Bus 1B-1.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	264000 K1.02	
Importance Rating	3.3	

K/A Statement: EMERGENCY GENERATORS (DIESEL/JET): Knowledge of the physical connections and/or cause-effect relationships between EMERGENCY GENERATORS (DIESEL/JET) and the following: D.C. electrical distribution (CFR 41.2 to 41.9)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	1. QCOA 6600-16, Unit 1 Diesel Generator Fails To Start, Rev. 03

Proposed references to be provided to applicants during examination: None

Learning Objective: SRN-6600-K23

Question Source:
Bank # X
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam Quad Cities Training Question Bank

Question Cognitive Level:
Memory or Fundamental Knowledge ___
Comprehension or Analysis X

Level of difficulty: (1-5) 2

10 CFR Part 55 Content:
55.41 2-9
55.43 ______

Comments:

Question: 52

Both units were operating at rated conditions when a Loss of ALL Offsite Power occurred on Unit 1 first and then Unit 2. The following Emergency Diesel Generator (EDG) responses are observed.

- Unit 1 EDG started and did not load.
- Unit 1/2 EDG started and loaded.
- Unit 2 EDG failed to start.

Which one of the following describes the expected power availability to the Instrument Air Compressors (IAC)?

	POWER to 1A IAC	POWER to 1/2 IAC	POWER to 1/2B IAC	POWER to 2 IAC
A.	Available	Available	Unavailable	Unavailable
B.	Unavailable	Unavailable	Available	Available
C.	Unavailable	Available	Unavailable	Unavailable
D.	Unavailable	Available	Available	Unavailable

Answer: C

Explanation:

A) Incorrect. Plausible because each of the listed air compressors is powered from a bus that can be supplied by an EDG. Power is lost to the 1A IAC when the Unit 1 EDG does not load.
B) Incorrect. Plausible because each of the listed air compressors is powered from a bus that can be supplied by an EDG. Power to the 2 IAC is lost when the 2 EDG failed to start and load. The Unit 1/2 EDG carries Bus 13-1 first since Unit 1 loses power first. Therefore, 1/2 IAC has power and not the 1/2B IAC.
C) Correct. Power is lost to the 1A IAC when the Unit 1 EDG does not load. Power to the 2 IAC is lost when the 2 EDG failed to start and load. The Unit 1/2 EDG carries Bus 13-1 first since Unit 1 loses power first. Therefore, 1/2 IAC has power and not the 1/2B IAC.
D) Incorrect. Plausible because each of the listed air compressors is powered from a bus that can be supplied by an EDG. The Unit 1/2 EDG carries Bus 13-1 first since Unit 1 loses power first. Therefore, 1/2 IAC has power but not the 1/2B IAC.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	300000K2.01	
Importance Rating	2.8	

K/A Statement: Instrument Air System (IAS): Knowledge of electrical power supplies to the following: Instrument air compressor (CFR: 41.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. LN-4701 Instrument Air, Rev. 13
2. LN-6600 Emergency Diesel Generator, Rev. 24
3. LN-6500 4kV/480 Distribution, Rev. 29

Proposed references to be provided to applicants during examination: None

Learning Objective: SRN-4701-K19 List the plant systems which support Instrument Air System and Describe the nature of support (Includes power supplies).

Question Source:

Bank #	
Modified Bank #	X (Note changes or attach parent)
New	

Question History: Last NRC Exam Monticello 2013 #50

Question Cognitive Level:	Memory or Fundamental Knowledge	____
	Comprehension or Analysis	X
Level of Difficulty:	(1-5)	2
10 CFR Part 55 Content:	55.41	7
	55.43	____

Comments:

Modified to include loss of power to both units and to use QC specific nomenclature.

Question: 53

One of the functions of the Reactor Building Closed Cooling Water system is to provide cooling water to….

A. the RWCU Regenerative Heat Exchangers to ensure system demineralizer resin is not damaged from overheating.

B. the RHR Heat Exchangers to remove heat from the Torus water during accident conditions and the reactor coolant system during shutdown cooling operations.

C. potentially contaminated components to minimize the release of radioactive material to the environment.

D. the Drywell coolers to ensure that the primary containment design temperature of 281ºF is not exceeded during accident conditions.

Answer: C

Explanation:

A) Incorrect. Plausible as RBCCW provides cooling water to the NON-Regenerative Heat Exchangers to ensure demineralizer resin isn't overheated.
B) Incorrect. Plausible as the examinee may believe the RBCCW system cools the RHR Heat Exchangers.
C) Correct. RBCCW cools certain Drywell, Reactor Building, and Radwaste loads. In addition, it is designed to operate at a lower pressure than its loads and service water which cools RBCCW in order to minimize the possibility of a release of radioactive material from a potentially contaminated system.
D) Incorrect. Plausible as the Drywell coolers are cooled by RBCCW. RBCCW pumps trip on a LOCA signal, and there are restrictions on restarting RBCCW to the Drywell with temperatures above 260 °F.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	400000 2.1.27	
Importance Rating	3.9	

K/A Statemen|t: Component Cooling Water: Knowledge of system purpose and/or function (CFR 41.7)

Technical Reference(s): LN-3700, Reactor Building Closed Cooling Water

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-3700-K01

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ____

Level of difficulty: (1-5) 2

10 CFR Part 55 Content:
55.41 7
55.43 _____

Comments:

Question: 54

Unit 1 Rod 30-29 inserts extremely slowly with the maximum procedurally allowed control rod drive water pressure and flow.

- All other control rods operate normally.
- No control rod blocks exist.
- Maintenance has been sent to troubleshoot the HCU.

Which of the following valves on the Control Rod Drive (CRD) Hydraulic Control Unit (HCU) is the most likely cause of this condition?

A. 1-0305-120 Directional Control Valve leaking

B. 1-0305-127 Scram Outlet Valve leaking

C. 1-0305-121 Directional Control Valve leaking

D. 1-0305-107 Accumulator Water Cylinder Drain Valve leaking

Answer: A

Explanation:

A) Correct. A leaking 1-0305-120 Directional Control Valve could allow drive water to leak into the exhaust header slowing rod insertion.
B) Incorrect. A leaking 1-0305-127 Scram Outlet Valve would allow depressurization of the under piston area potentially causing the rod to drift in.
C) Incorrect. A leaking 1-0305-121 Directional Control Valve could allow drive water to leak during rod withdrawal slowing rod out motion. This valve is opened during rod insertion.
D) Incorrect. A leaking 1-0305-107 Accumulator Water Cylinder Drain Valve would divert some drive water to maintain scram accumulator pressure and would affect other control rod drive speeds.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	201001K3.03	
Importance Rating	3.1	

K/A Statement: Control Rod Drive Hydraulic System: Knowledge of the effect that a loss or malfunction of the CONTROL ROD DRIVE HYDRAULIC SYSTEM will have on following: Control Rod Drive Mechanisms (CFR: 41.7 / 45.4)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
1. LIC-0302 CRD Hydraulics, Rev. 15

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0302-K22g Given a Control Rod Drive Hydraulics operating mode and various plant conditons, PREDICT how key system/plant parameters will respond to the following Control Rod Drive Hydraulics component or controller failures. g.) Leaking Directional Control Valves (insert or withdrawal).

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
Level of Difficulty:	(1-5)	2
10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:

Question: 55

Unit 1 was at 58% power when an automatic reactor SCRAM was received due to a main turbine trip.

- Multiple control rods FAILED to fully insert into the reactor core
- Reactor power is 7% and steady
- Control rod J-6 is selected and is currently at position 24
- Control rod J-6 is NOT in the current sequence step
- The RWM mode switch is in NORMAL

1.) Control rod J-6 Full Core Display rod position indication will be _________.

2.) If the ROD OUT NOTCH OVERRIDE switch is held in the EMERG ROD IN position, control rod J-6 Full Core Display rod position indication will _________.

A. (1) GREEN 24
(2) change

B. (1) AMBER 24
(2) change

C. (1) AMBER 24
(2) remain the same

D. (1) GREEN 24
(2) remain the same

Answer: __C__

Explanation:

Control position indication color will either be GREEN (full in), AMBER (neither full in nor full out), or RED (full out).

A) Incorrect. J-6 position indication will be AMBER not GREEN; plausible since GREEN is one of the three colors used in displaying control rod position. The RWM will block the out of sequence J-6 and it will remain at AMBER 24. Plausible since the control rod would be expected to insert if not for the RWM block.
B) Incorrect. J-6 position indication will be AMBER. The RWM will block the out of sequence J-6 and it will remain at AMBER 24. Plausible since the control rod would be expected to insert if not for the RWM block.
C) Correct. J-6 position indication will be AMBER. The ROD OUT NOTCH OVERRIDE switch held in the EMERG ROD IN will not override the RWM insert block on out of sequence.
D) Incorrect. J-6 position indication will be AMBER not GREEN; plausible since GREEN is one of the three colors used in displaying control rod position. The ROD OUT NOTCH OVERRIDE switch held in the EMERG ROD IN will not override the RWM insert block on out of sequence.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201003 A3.01	
	Importance Rating	3.7	

K/A Statement: CONTROL ROD AND DRIVE MECHANISM: Ability to monitor automatic operations of the CONTROL ROD AND DRIVE MECHANISM including: Control rod position (CFR 41.7)

Technical Reference(s):
(Attach if not previously provided, including version/revision number.)

1. QCOA 0280-01, RPIS Failure, Rev. 17
2. LIC 301, Control Rod Blade and Drive Mechanism
3. LIC 0280, Reactor Manual Control and Rod Position Information Systems

Proposed references to be provided to applicants during examination: ___None__

Learning Objective: SR-0280-K20

Question Source:
Bank # _______
Modified Bank # __X___ (Parent attached)
New ___–___

Question History: Last NRC Exam Quad Cities NRC ILT Exam 2011_____

Question Cognitive Level:	Memory or Fundamental Knowledge	___
	Comprehension or Analysis	X
Level of difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	7
	55.43	___

Comments:

Question modified to test knowledge of control rod position indication instead of knowledge related to Sequence Timer.

Question: 56

The earliest that the Rod Worth Minimizer (RWM) can be bypassed, without implementing compensatory actions, is when __(1)___ because protection against a design basis __(2)___ is no longer required.

A. (1) reactor power is > 10% RTP
   (2) Control Rod Drop Accident

B. (1) reactor power is > 10% RTP
   (2) Rod Withdrawal Error

C. (1) the reactor enters Mode 1
   (2) Control Rod Drop Accident

D. (1) the reactor enters Mode 1
   (2) Rod Withdrawal Error

Answer: A

Explanation:

A) Correct. (1) TS 3.3.2.1 table for control rod block instrumentation only requires the RWM to be operable at < or = 10% RTP. (2) The purpose of the RWM is to help protect against a control rod drop accident.
B) Incorrect. (1) TS 3.3.2.1 table for control rod block instrumentation only requires the RWM to be operable at < or = 10% RTP. (2) The Rod Block Monitor, NOT the RWM protects against the negative effects of a Rod Withdrawal Error per TS 3.3.2.1 bases. The TS 3.3.2.1 bases also discuss the purpose of the RWM.
C) Incorrect. (1) The RWM may be bypassed when power exceeds 10% RTP. Mode 1 entry may occurs when >5% RTP. (2) This portion is correct. The purpose of the RWM is to help protect against a control rod drop accident.
D) Incorrect. (1) The RWM may be bypassed when power exceeds 10% RTP. Mode 1 entry may occurs when >5% RTP. (2) The Rod Block Monitor, NOT the RWM, protects against the negative effects of a Rod Withdrawal Error per TS 3.3.2.1 bases. The TS 3.3.2.1 bases also discuss the purpose of the RWM.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	201006K1.05	
Importance Rating	3.5	

K/A Statement: Rod Worth Minimizer System (RWM): Knowledge of the physical connections and/or cause effect relationships between ROD WORTH MINIMIZER SYSTEM (RWM) and the following: Control rod drop accident (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. LIC-0207 Rod Worth Minimizer, Rev. 14
2. TS 3.3.2.1 and Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0207-K01 STATE the purpose(s) of the Rod Worth Minimizer including applicable design bases.

Question Source:
Bank # ________
Modified Bank # ________ (Note changes or attach parent)
New ___X___

Question History: Last NRC Exam __N/A______

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
Level of Difficulty:	(1-5)	2
10 CFR Part 55 Content:	55.41	6 and 7
	55.43	

Comments:

RO Level justification. The first part of the question asks when the RWM portion of TS 3.3.2.1 is applicable. This portion is RO level of knowledge. The question also asks about information that is in the bases for TS 3.3.2.1. The licensee has an RO level learning objective that requires knowledge of the accident that is protected against with the Rod Worth Minimizer.

Question: 57

Both Units are at full power.

<u>1(2)-1290-28 RWCU Loop Temperature Select</u>	<u>Unit 1</u>	<u>Unit 2</u>
Point 1 WATER TO RWCU SYSTEM	540°F	540°F
Point 2 REGEN HX OUTLET	230°F	200°F
Point 3 NON-REGEN HX OUTLET	130°F	100°F
Point 4 RETURN TO FEEDWATER LINE	410°F	440°F

Which plant condition could be causing the differences in temperature in the Unit 1 and Unit 2 RWCU systems?

A. Failing RBCCW Heat Exchanger temperature controller TIC 1-3941-21A

B. 1-1299-78, Regenerative Heat Exchanger Bypass Valve is partially OPEN

C. 1-1204-B set of Non-Regenerative Heat Exchangers are isolated

D. MO 1-1201-133 RWCU Filter/Demineralize Bypass Valve is throttled OPEN

Answer: B

Explanation:

A) Incorrect. Plausible as this temperature controller failing could cause higher RBCCW temperatures, which will in turn cause NRHX outlet temp (PT 3) to go up, but it will also result in the RWCU water returning to the feedwater system (PT 4) going up.

B) Correct. The RHX Bypass Valve partially open means some reactor water bypasses heat exchanger when entering the RWCU system. Less heat will be removed from the RHX. This will cause NRHX inlet and outlet temperatures to be higher. In addition, it will mean that the RWCU water returning to the feedwater system will not be heated up as much (loss of efficiency) and therefore the RWCU return to feedwater temperature will be less.

C) Incorrect. With a train of the NRHX offline, flow rate through the RWCU will be reduced by approximately half. This will mean the NRHX inlet temperature will be much higher, but the outlet temperature may be somewhat higher but the heat transfer to the RBCCW system will be higher also. The RWCU water returning to the feedwater system temperature would be lower since less heat was added by the RHX.

D) Incorrect. This valve only bypasses the filter/demineralizers and has no effect on system temperatures as read on TR 1(2)-1290-21 at the 901(2)-4 panels.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	204000 K4.06	
Importance Rating	2.6	

K/A Statement: REACTOR WATER CLEANUP SYSTEM: Knowledge of REACTOR WATER CLEANUP SYSTEM design feature(s) and/or interlocks which provide for the following: Maximize plant efficiency (use of regenerative heat exchanger) (CFR 41.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
1. LN-1200, Reactor Water Cleanup

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-1200-K20c

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	___
	Comprehension or Analysis	_X_
Level of difficulty:	(1-5) __2__	
10 CFR Part 55 Content:	55.41 __7__	
	55.43 _____	

Comments:

Question: 58

Unit 1 is operating at 100% power.

- Traversing In-Core Probe (TIP) scans are in progress with TIP Drive 1 detector in the core.
- A reactor coolant leak develops in the drywell.
- Drywell pressure is 3 psig and rising.
- TIP Drive 1 detector fails to retract automatically and manually.

The TIP Drive 1 shear valve…

A. must be closed, locally, using a keylock switch located adjacent to the TIP drive mechanisms.

B. must be closed using a keylock switch located on Control Room Panel 901-13.

C. automatically closes if the detector is not returned to its index mechanism, within the prescribed time delay, following the PCIS Group 2 actuation.

D. automatically closes if the detector is not returned to its shield chamber, within the prescribed time delay, following the PCIS Group 2 actuation.

Answer: B

Explanation:

A) Incorrect. The TIP shear valve closure requires operator action, but there is not a local switch to operate the shear valve.
B) Correct. The TIP shear valves are provided as a backup to the normal primary containment isolation feature. No automatic shear valve actuation is provided. The operator must detect the failure and manually close the shear valve from Control Room Panel 901-13.
C) Incorrect. The TIP drive control unit will automatically retract the TIP to the shield chamber and close the ball valve on a Group II primary containment isolation signal, but there is no automatic operation feature for the shear valve.
D) Incorrect. The TIP drive control unit will automatically retract the TIP to the index mechanism upon completion of a scan, but on a Group II primary containment isolation signal the detector is retracted to the shield chamber. There is no automatic operation feature for the shear valve.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	215001K4.01	
Importance Rating	3.4	

K/A Statement: Traversing In-Core Probe: Knowledge of TRAVERSING IN-CORE PROBE design feature(s) and/or interlocks which provide for the following: Primary containment isolation (CFR: 41.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. LIC-0704B Traversing In-Core Probe (Unit 1), Rev. 13
2. LIC-0704A Traversing In-Core Probe (Unit 2), Rev. 12
3. QCOP 0700-08 Unit 1 TIP, Rev. 18

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0704-K26 Evaluate given key TIP System 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): e. Failure to retract

Question Source:
Bank # X
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam Fitzpatrick 2012 #57

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	____
Level of Difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	7
	55.43	_____

Comments: Changed the stem and answers to match Quad Cities specific terminology.

Question: 59

Each Reactor Vessel Level instrument channel Master Trip Unit (MTU)/Slave Trip Unit (STU) have “GROSS FAIL TRIP” and “TRIP” LEDs.

When the condition causing the “tripped” status clears:

1) The GROSS FAIL TRIP circuit _________.
2) The TRIP circuit _________.

A. 1) resets automatically
   2) resets automatically

B. 1) must be manually reset
   2) resets automatically

C. 1) resets automatically
   2) must be reset manually

D. 1) must be reset manually
   2) must be reset manually

Answer: B

Explanation:

Each master trip unit and slave trip unit has a gross failure trip which will latch if the signal output from the trip unit is outside prescribed bands. In order to reset the gross failure for the affected trip unit, the condition that caused the gross failure must be corrected, then the gross failure reset push-button on the front of the trip unit will need to be depressed. This will reset the gross failure and gross failure LED.

When a trip occurs, the related trip light will illuminate and will remain illuminated as long as the sensed parameter exceeds the trip setpoint. When the sensed parameter returns to normal (falls within trip reset value), the trip unit's trip output automatically resets and the trip light will extinguish.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	216000 A1.03	
Importance Rating	2.9	

K/A Statement: NUCLEAR BOILER INSTRUMENTATION: Ability to predict and/or monitor changes in parameters associated with operating the NUCLEAR BOILER INSTRUMENTATION controls including: Surveillance testing (CFR 41.5)

Technical Reference(s): LIC-0263, Reactor Vessel Instrumentation
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: SRN-0263-K14

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge ___
Comprehension or Analysis X

Level of difficulty: (1-5) 2

10 CFR Part 55 Content:
55.41 5
55.43 _____

Comments:

Question: 60

A small break LOCA has occurred on Unit 1.

QGA 100, Reactor Pressure Vessel Control and QGA 200, Primary Containment Control are being implemented.

RHR is being operated with Torus Cooling and Torus Sprays on RHR Loop A when RHR Pump C trips.

Which of the following describes the use of QCOA 1000-03, RHR Pump Trip under these conditions?

A. Entry into and implementation of QCOA 1000-03 is not required.

B. Entry into QCOA 1000-03 RHR Pump Trip is required, but implementation is deferred until after the QGAs are exited.

C. Enter and implement the actions specified in QCOA 1000-03 concurrently with the QGAs. If procedure conflicts arise, the actions specified by the QGAs have precedence.

D. Enter and implement the actions specified in QCOA 1000-03 concurrently with the QGAs. If procedure conflicts arise, the actions specified by the QCOA have precedence.

Answer: C

Explanation:

A) Incorrect. Procedures are to be performed concurrently with the caveat that QGA actions have precedence. Plausible since QGAs take precedence over QCOAs.
B) Incorrect. The QCOA procedure is to be performed concurrently with the QGA procedures, but QGA actions take precedence. The QCOA may provide guidance that will facilitate implementation of the QGAs.
C) Correct. The QCOA procedure is to be performed concurrently with the QGA procedures, but QGA actions take precedence.
D) Incorrect. QGA actions take precedence over QCOA actions.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	230000G2.4.8	
	Importance Rating	3.8	

K/A Statement: RHR/LPCI: Torus/Suppression Pool Spray Mode: Knowledge of how abnormal operating procedures are used in conjunction with EOPs. (CFR: 41.10 / 43.5 / 45.13)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. LN-1000 RHR system, Rev. 20
2. QCOP 1000-30 Post-Accident RHR Operation, Rev. 31
3. QCOA 1000-03 RHR Pump Trip, Rev. 10
4. LN-PROC Procedures, Rev. 15

Proposed references to be provided to applicants during examination: None

Learning Objective: SRNLF-PR-K01 DESCRIBE the Exelon Nuclear procedure hierarchy including the relationship between:

- b. Station Procedures
  - 2. Operating Procedures
  - 4. Annunciator Procedures
  - 5. Emergency Operating Procedures
  - 8. Operating Abnormal Procedures

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam D.C. Cook 2010 #16

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
Level of Difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	10
	55.43	

Comments:

Modified the question to fit the K/A; and Quad Cities terminology and technology.

Question: 61

Unit 1 is at 100% rated thermal power.

- Annunciator 901-7, B-6, EHC FLUID RESERVOIR LEVEL is in alarm.
- The Turbine Building operator reports that the EHC Reservoir level has lowered approximately 1” over the last 8 hours.
- Turbine Control Valve (TCV) #2 accumulator is determined to be leaking EHC fluid

Which one of the following describes the expected plant and operator responses to the above event?

A. Trip of the EHC pump is imminent; Insert a Manual SCRAM and Manually Trip the Main Turbine.

B. TCV #2 will fail closed due to low pressure as its accumulator drains; Perform an Emergency Power Reduction to < 70% RTP.

C. The oil supply to TCV #2 must be isolated to stop the loss of fluid from the reservoir; Perform a controlled power reduction to < 70% RTP to remove TCV #2 from service.

D. The EHC system must be removed from service to stop the loss of fluid from the reservoir. Perform a normal plant shutdown to permit shutdown of the EHC system to make repairs.

Answer: C

Explanation:

A) Incorrect. There are no EHC pump trips associated with low level in the EHC reservoir. The leak rate is slow enough to permit a controlled response to the leak.
B) Incorrect. The valve would fail closed only if there were a catastrophic failure of the supply to the TCV or system wide loss of pressure. Power must be reduced to less than 70% RTP to remove a TCV from service.
C) Correct. Leak is small enough to permit a controlled power reduction to < 70% power so that the TCV can be isolated. Power operation may continue until the next outage if necessary with one TCV isolated.
D) Incorrect. An EHC leak on a TCV accumulator can be addressed by isolating the accumulator without shutdown of the EHC system; therefore a plant shutdown is not required. With one TCV accumulator isolated, plant operation is permitted until the next outage.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	241000 A2.20	
Importance Rating	2.5	

K/A Statement: REACTOR/TURBINE PRESSURE REGULATING SYSTEM: Ability to (a) predict the impacts of the following on the REACTOR/TURBINE PRESSURE REGULATING SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low reactor/turbine pressure regulating system oil level: plant specific (CFR 41.5)

Technical Reference(s): Attach if not previously provided, including version/revision number.)

1. QCAN 901-7, B-6, EHC FLUID RESERVOIR LEVEL; Rev. 7
2. QCOP 5600-07 Isolating and Unisolating One Main Turbine Valve; Rev 17
3. QCGP 2-3, Reactor SCRAM; Rev 87
4. QCGP 3-1, Reactor Power Operations; Rev 85

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-5650-K20 and SRN-5650-K14

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	____
	Comprehension or Analysis	_X_
Level of difficulty:	(1-5) __3__	
10 CFR Part 55 Content:	55.41 __5__	
	55.43 _____	

Comments:

Question: 62

Unit 1 is critical and heating up.

- Reactor pressure 85 psig and rising.
- Main turbine on the turning gear.
- Gland Seal Steam Supply pressure is 6.5 psig

(1) What is the operational concern with the Gland Seal Steam Supply pressure?

(2) What operator action is required to return the Gland Seal Steam Supply pressure to its normal band?

A. (1) Gland seal supply header relief valves are about to lift

(2) Throttle shut MO 1-3099-S1, STM SEAL FEED VLV

B. (1) Gland seal supply header relief valves are about to lift

(2) Throttle shut MO 1-3099-S2, STM SEAL BYP FEED VLV

C. (1) Water intrusion into the turbine oil

(2) Throttle shut MO 1-3099-S1, STM SEAL FEED VLV

D. (1) Water intrusion into the turbine oil

(2) Throttle shut MO 1-3099-S2, STM SEAL BYP FEED VLV

Answer: D

Explanation:

A) Incorrect. Plausible because the gland seal supply header relief valves lift at 20 psig, but there is no information in the stem to indicate that gland seal pressure is rising. With reactor pressure less than 100 psig, MO 1-3099-S2 will have been throttled open to assist (raise pressure into the control band) PCV 1-3099-124 in maintaining Gland Seal Supply pressure. With Gland Seal Pressure at 6.5 psig the PCV should also be shut, negating any need to throttle S1 shut. Plausible since throttling shut S1 would be required in the PCV failed open.
B) Incorrect. Plausible because the gland seal supply header relief valves lift at 20 psig, but there is no information in the stem to indicate that gland seal pressure is rising. Throttling S2 shut is the correct action.
C) Incorrect. Too high of a gland seal steam supply pressure could cause water intrusion into the turbine oil. 2.5 to 5.0 psig is the normal pressure range of the gland seal steam supply pressure. With reactor pressure less than 100 psig, MO 1-3099-S2 will have been throttled open to assist (raise pressure into the control band) PCV 1-3099-124 in maintaining Gland Seal Supply pressure. With Gland Seal Pressure at 6.5 psig the PCV should also be shut, negating any need to throttle S1 shut. Plausible since throttling shut S1 would be required in the PCV failed open.
D) Correct. Too high of a gland seal steam supply pressure could cause water intrusion into the turbine oil. Throttling S2 shut is the correct action.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	245000K5.06	
	Importance Rating	2.5	

K/A Statement: Main Turbine Generator and Auxiliary Systems: Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS: Turbine shaft sealing (CFR: 41.5 / 45.3)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. LIC-5600 Main Turbine and Auxiliaries, Rev. 19
2. QOP 5600-01 Gland Seal System, Rev. 23

Proposed references to be provided to applicants during examination: N/A

Learning Objective: SR-5600-K22
a.) Gland seal pressure regulator fails high/low
f.) Gland Steam Exhauster trips

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
Level of Difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	5
	55.43	

Comments:

Question: 63

A reactor startup is in progress on Unit 1.

- Reactor pressure is 300 psig
- ‘B’ and ‘C’ condensate/condensate booster pumps are running
- ‘C’ reactor feedwater pump was just started per QOP 3200-02
- COND PMP SELECTOR switch positioned to ‘PUMP 1A’
- RFP SELECTOR switch positioned to ‘OFF’

At 0800:00, annunciator 901-5, H-7 RX FEED PUMP SUCTION LO/LO PRESSURE alarms.

Time	RFP Suction Pressure (psig)
0800:00	145
0800:03	135
0800:05	124
0800:11	250

With NO operator actions, what will be the expected condition of the condensate and feedwater system?

A. Auto trip of the ‘C’ reactor feedwater pump ONLY

B. Auto start of the ‘A’ condensate/condensate booster pump ONLY

C. Auto start of the ‘A’ condensate/condensate booster pump; AND auto trip of the ‘C’ reactor feedwater pump ONLY

D. Auto start of the ‘A’ condensate/condensate booster pump; auto trip of the ‘C’ reactor feedwater pump; AND auto start of the standby reactor feedwater pump

Answer: C

Explanation:

A) Incorrect. The ‘C’ RFP will trip on low RFP suction pressure < 125 psig for 5 seconds. Standby ‘A’ CD/CB would have auto started on low RFP suction pressure < 145 psig.
B) Incorrect. The ‘A’ CD/CB will auto start on low RFP suction pressure < 145 psig and the ‘C’ RFP would have tripped on low RFP suction pressure < 125 psig for 5 seconds.
C) Correct. The standby ‘A’ CD/CB pump will auto start when RFP suction pressure < 145 psig and the ‘C’ RFP auto trips on RFP suction pressure < 125 psig for 5 seconds.
D) Incorrect. The ‘A’ CD/CB pump will auto start and the ‘C’ RFP will trip after suction pressure is < 125 psig for 5 seconds. A standby RFP will not start since the RFP Selector in OFF.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	256000 K6.06	
Importance Rating	3.3	

K/A Statement: REACTOR CONDENSATE SYSTEM: Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR CONDENSATE SYSTEM: Reactor feedwater system (CFR 41.7)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QCGP 1-1, Normal Unit 1 Startup, Rev. 110
2. QOP 3200-02, Start Up of the First Reactor Feed Pump, Rev. 42
3. LIC-3200, Feed and Condensate

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-3200-K22

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge ___
Comprehension or Analysis X

Level of difficulty: (1-5) 3

10 CFR Part 55 Content:
55.41 7
55.43 ______

Comments:

Question: 64

The Unit 1 Drywell Floor Drain Sump (DWFDS) flowrate indication is out of service. The integrator still works.

The previous 24 hour unidentified leakage rate was 0.13 gpm and stable for each 4-hour period.

The volume of water pumped from the Drywell Floor Drain Sump (DWFDS) at four hour intervals for the last 24 hours was recorded as follows:

0000 – 0400	32 gallons
0400 – 0800	151 gallons
0800 - 1200	400 gallons
1200 – 1600	497 gallons
1600 – 2000	1119 gallons
2000 – 2400	1499 gallons

The RCS Operational Leakage LCO was first exceeded at ….

A. 1200

B. 1600

C. 2000

D. 2400

Answer: C

Explanation:

A) Incorrect: Plausible because QCOS 1600-07 requires notification of the Station Duty Manager when unidentified leakage reaches >240 gallons change in a 4 hour period (>1 gpm) , leakage is still less than the LCO limit.
B) Incorrect: Plausible because the leakage rate (2.07 gpm) exceeds 2 gpm in a 4 hour period but has only increased by 0.4 gpm over the previous 4 hour interval (1.67 gpm).
C) Correct: The leakage rate (4.66 gpm) has increased by >2 gpm over the previous 4 hour interval as well as being >2 gpm above the previous 24 hour average of 0.13 gpm.
D) Incorrect: Plausible because the leakage rate (6.25 gpm) exceeds the 5 gpm LCO limit for total leakage but is not the first LCO limit to be exceeded.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	268000A4.01	
Importance Rating	3.4	

K/A Statement: Radwaste: Ability to manually operate and/or monitor in the control room: Sump integrators (CFR: 41.7 / 45.5 to 45.8)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
1. LN-2000 Radioactive Waste Processing, Rev. 17
2. QCOS 1600-07 Reactor Coolant System Leakage in the Drywell (DWFDS and DWEDS Available)

Proposed references to be provided to applicants during examination: QCOS 1600-07

Learning Objective: SR-2000-K20 Given a Radwaste Liquid Processing System operation and various plant conditions, EVALUATE the following Radwaste Liquid Processing System indications/responses and DETERMINE if the indication/ response is expected and normal.
d. DWEDS/DWFDS flow integrators

Question Source:
Bank # ________
Modified Bank # ________ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis x

Level of Difficulty: (1-5) 3

10 CFR Part 55 Content:
55.41 7
55.43 _____

Comments:

Question: 65

A loss of MCC 16-2 will result in a loss of the power supply to the ….

A. 1/2A Diesel Fire Pump battery charger.

B. 1/2A Diesel Fire Pump starting motor.

C. 1/2B Diesel Fire Pump battery charger.

D. 1/2B Diesel Fire Pump starting motor.

Answer: A

Explanation:

A) Correct. A loss of MCC 16-2 will result in a loss of power supply for the battery charger associated with the 4101A DFP. The battery will be able to start the 4101A on a low pressure signal and an engine mounted generator would charge the starting battery during engine operation.
B) Incorrect. Plausible as the MCC 16-2 powers the battery charger for the 4101A DFP. The starting motor for the 4101A DFP receives power directly from its starting battery.
C) Incorrect. MCC 26-2 powers the battery charger associated with the 4101B DFP.
D) Incorrect. MCC 26-2 powers the battery charger for the 4101B DFP. The starting motor for the 4101B DFP receives power directly from its starting battery.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	286000 K2.02	
Importance Rating	2.9	

K/A Statement: FIRE PROTECTION SYSTEM: Knowledge of electrical power supplies to the following: Pumps (CFR 41.7)

Technical Reference(s): LN-4100, Fire Protection System
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-4100-K15

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ____

Level of difficulty: (1-5) 2

10 CFR Part 55 Content:
55.41 7
55.43 _____

Comments:

Question: 66

Which of the following personnel can operate the controls at Quad Cities Nuclear Power Station (QCNPS) supervised by a licensed reactor operator?

- John who applied for, but has not yet been accepted into the next QCNPS licensed operator training class
- Mary who is licensed as a Reactor Operator at Dresden Generating Station (DGS)
- Sally who is a Senior Reactor Operator trainee in the current licensed operator training class at QCNPS
- Ted who was a licensed Reactor Operator at QCNPS and is reactivating his license
- Steve who is a Senior Reactor Operator trainee in the current licensed operator training class at DGS

A. Sally and Steve ONLY

B. Sally and Ted ONLY

C. Mary, Sally, Ted, and Steve ONLY

D. John, Sally, and Ted ONLY

Answer: B

Explanation:

A) Incorrect. Plausible because both Sally and Steve are currently enrolled in a certified training program. However, Steve is not enrolled in a training program at QCGS.
B) Correct. Sally is enrolled in a training program at QCGS. Ted previously had a QCGS RO license, but because he is reactivating it, he must be supervised by a Reactor Operator with an active license. The process of reactivating a license is covered by a QCGS training procedure.
C) Incorrect. Plausible because Mary has an active license but is not licensed at QCGS. Steve is enrolled in an approved training program, but not at QCGS. Sally and Ted would be allowed to operate the controls under the supervision of a licensed RO at QCGS.
D) Incorrect. Plausible because John, Sally, and Ted are all QCGS employees. However, it is incorrect because John is not yet enrolled in an approved training program at QCGS.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	
Group #	-	
K/A #	G2.1.37	
Importance Rating	4.3	

K/A Statement: Knowledge of procedures, guidelines, or limitations associated with reactivity management. (CFR: 41.1 / 43.6 / 45.6)

Technical Reference(s): OP-AA-300, Reactivity Management, Rev. 11
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: N/A

Learning Objective: SR-PGRM-K1 (Freq: LIC=A) Given OP-AA-300, Reactivity Management, DESCRIBE the responsibilities of the different job positions required to complete the procedure. (i.e., Who does what?)

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis X

Level of Difficulty: (1-5) 3

10 CFR Part 55 Content:
55.41 1
55.43 ______

Comments:

Question: 67

Shift turnover is in progress on Unit 1. Reactor instantaneous power is 2957 MWth and slowly RISING, due to Xenon burn-off. The off-going Unit Supervisor directs the off-going NSO to lower power using Reactor Recirculation pumps.

Which of the following describes the station's policy for completing this power change during the shift turnover?

A. The power change should be delayed until the Shift Turnover is complete and on-coming personnel have assumed the watch.

B. The Unit Supervisor should place the Shift Turnover on hold; the off-going NSO should make the power change; then the Shift Turnover would be resumed.

C. The on-coming NSO should assume the At-the-Controls NSO duties and make the power change while other Main Control Room personnel continue their turnover activities during the power change.

D. The off-going NSO should put his turnover on hold; make the power change; then resume his turnover. Other Main Control room personnel may continue their turnover activities during the power change.

Answer: B

Explanation:

A) Incorrect. OP-AA-300 directs the Unit Supervisor to minimize control room distractions during reactivity maneuvers including turnover.
B) Correct. Conservative decision making dictates that the operator who has been at-the-controls and has been addressing the ongoing xenon transient should complete the short duration reactivity maneuver prior to shift turnover.
C) Incorrect. Plausible as this outcome is not strictly prohibited, but it does not represent conservative decision making.
D) Incorrect. Plausible as the decision to have the offgoing operator perform the reactivity maneuver is the operating with conservative decision fundamentals in mind.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	
Group #	1	
K/A #	2.1.39	
Importance Rating	3.6	

K/A Statement: Knowledge of conservative decision making practices. (CFR 41.10)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
1. OP-AA-300, Reactivity Management, Rev 11
2. OP-AA-112-101, Shift Turnover and Relief, Rev. 12
3. LIC-QCGP, QCGP Procedures

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0002-K06 and SR-CROP-K10

Question Source:
Bank # X
Modified Bank # ______ (Note changes or attach parent)
New _

Question History: Last NRC Exam Grand Gulf NRC ILT Exam 2009

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ____

Level of difficulty: (1-5) 2

10 CFR Part 55 Content:
55.41 10
55.43 _____

Comments:

Question: 68

The reactor is in Mode 4. Control rod exercising is in progress in preparation for a reactor startup.

(1) The Reactor Mode Switch is in the __________ position.
(2) A Qualified Verifier (QV) __________ required for each control rod maneuver during the control rod exercise.

A. (1) STARTUP
   (2) is

B. (1) STARTUP
   (2) is NOT

C. (1) REFUEL
   (2) is

D. (2) REFUEL
   (2) is NOT

Answer: __D__

Explanation:

A) Incorrect. (1) Plausible since the mode switch must be taken out of SHUTDOWN to with draw control rods. TS LCO 3.10.3 permits single control rod withdrawal in Mode 4 provided that the Reactor Mode switch is in REFUEL so that withdrawal of more than one control rod is prevented. With the Reactor Mode Switch in STARTUP, there would be no interlock to prevent withdrawal of more than one control rod. (2) Exelon corporate procedures require peer checks for rod motion only in Modes 1 and 2. Plausible since use of peer checks is a normal expectation for movement of control rods.
B) Incorrect. (1) Plausible since the mode switch must be taken out of SHUTDOWN to with draw control rods. TS LCO 3.10.3 permits single control rod withdrawal in Mode 4 provided that the Reactor Mode switch is in REFUEL so that withdrawal of more than one control rod is prevented. With the Reactor Mode Switch in STARTUP, there would be no interlock to prevent withdrawal of more than one control rod. (2) Exelon corporate procedures require peer checks for rod motion only in Modes 1 and 2.
C) Incorrect. (1) The mode switch must be taken out of SHUTDOWN to with draw control rods. TS LCO 3.10.3 permits single control rod withdrawal in Mode 4 provided that the Reactor Mode switch is in REFUEL so that withdrawal of more than one control rod is prevented. (2) Exelon corporate procedures require peer checks for rod motion only in Modes 1 and 2. Plausible since use of peer checks is a normal expectation for movement of control rods.
D) Correct. (1) The mode switch must be taken out of SHUTDOWN to with draw control rods. TS LCO 3.10.3 permits single control rod withdrawal in Mode 4 provided that the Reactor Mode switch is in REFUEL so that withdrawal of more than one control rod is prevented. (2) Exelon corporate procedures require peer checks for rod motion only in Modes 1 and 2.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	____
Group #	____	____
K/A #	G2.2.1	
Importance Rating	4.5	____

K/A Statement: Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.
(CFR: 41.5 / 41.10 / 43.5 / 43.6 / 45.1)

Technical Reference(s):
(Attach if not previously provided, including version/revision number.)

1. OP-AB-300-1001 BWR Control Rod Movement Requirements, Rev. 9
2. OP-AB-300-1005 BWR Reactivity Management – Shutdown Activities, Rev. 7
3. QCOP 0300-18 Control Rod Exercising, Rev. 34

Proposed references to be provided to applicants during examination: __None__

Learning Objective: SR-PGRM-K2 DESCRIBE the General Requirements for reactivity changes in accordance with OP-AA-300, Reactivity Management.

Question Source:	Bank #	________	
	Modified Bank #	________	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis	____	
Level of Difficulty:	(1-5)	3	
10 CFR Part 55 Content:	55.41	_____	
	55.43	_____	

Comments:

Question: 69

A Clearance Order is being written to deenergize MCC-30 to perform breaker inspections following a minor fire. Currently both Units are operating at full power.

Who must approve the Clearance Order Checklist and why?

A. Only an active SRO can approve the checklist because MCC-30 has feeds from both units' safety-related busses

B. A licensed (or previously licensed) individual must approve the checklist because the equipment powered from MCC-30 is Tech Spec related

C. The Shift Manager must approve the checklist because deenergizing MCC-30 requires both Unit 1 and Unit 2 to enter Administrative Technical Requirements (ATR)

D. The Shift Manager must approve the checklist because deenergizing MCC-30 requires both Unit 1 and Unit 2 to enter Tech Spec LCOs

Answer: B

Explanation:

A) Incorrect. Plausible since this is an administrative task that are normally associated with the SROs, but OP-AA-109-101 would allow an RO to approve the checlist.
B) Correct. OP-AA-109-101, “Clearance and Tagging” states that a licensed (or previously licensed) individual will approve the checklist if the C/O is Tech Spec or Safety Related. MCC-30 powers SSMP systems and is therefore Tech Spec related.
C) Incorrect. Plausible as Shift Manager’s approval is not required for the clearance order, but is for work authorization.
D) Incorrect. Plausible as Shift Manager’s approval is not required for the clearance order, but is for work authorization.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	
Group #	1	
K/A #	2.2.13	
Importance Rating	4.1	

K/A Statement: Knowledge of tagging and clearance procedures (CFR 41.10)

Technical Reference(s): OP-AA-109-101, Clearance and Tagging, Rev. 12
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: SRNL-CO-K4

Question Source:
Bank # X
Modified Bank # ______ (Note changes or attach parent)
New _

Question History: Last NRC Exam Quad Cities Written Exam Bank

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ____

Level of difficulty: (1-5) 3

10 CFR Part 55 Content:
55.41 10
55.43 _____

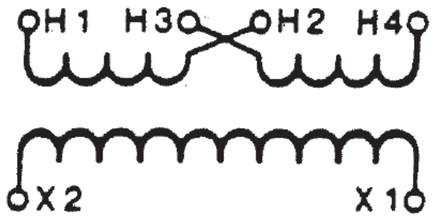
Comments:

ID: 340657

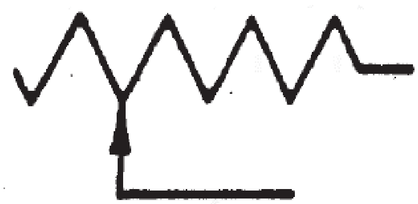
Question: 70

Which one of the following symbols represents a tapped resistor in standard Sargent and Lundy electrical drawings?

A)



B)



C)

D)

Answer: B

Explanation:

A) Incorrect. This is the symbol for a control transformer.
B) Correct. This is the symbol for a tapped resistor.
C) Incorrect. This is the symbol for a fixed resistor
D) Incorrect. This is the symbol for a fuse.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	____
Group #	____	____
K/A #	G2.2.41	
Importance Rating	4.5	___

K/A Statement: Ability to obtain and interpret station electrical and mechanical drawings. (CFR: 41.10 / 45.12 / 45.13)

Technical Reference(s): LN EPRN Electrical Prints, Rev. 13
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: SRN-EPR-K02 DESCRIBE the basic format and standard symbols on Electrical Drawings and representation of typical components such as switches, relays, contacts, etc...

Question Source:
Bank # ________
Modified Bank # ________ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ______

Level of Difficulty: (1-5) 2

10 CFR Part 55 Content:
55.41 10
55.43 _____

Comments:
I am considering this new because I wrote a similar question for an exam that will be given at nearly the same timeframe as Quad Cities. The Quad Cities exam will be given approximately one month after the other exam. If you want to change this to a modified question, that would be fine.

Question: 71

A fire has developed in an area where radiation levels are 10 R/hr. One of the fire-brigade members has been injured and could not be removed from the area. Because of the nature and progression of the fire, his life is in imminent danger. The Shift Manager, acting as the Station Emergency Director, has authorized entry into the area to rescue the injured Operator

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.

Using the highest permissible dose limit for these conditions, what is the maximum amount of time that the rescuer can be in the area?

A. 12 minutes

B 30 minutes

C. 60 minutes

D. There is no limit

Answer: D

Explanation:

A) Incorrect, Plausible because 2 rem is the Exelon administrative TEDE limit.
B) Incorrect, Plausible because 5 REM is the federal TEDE limit.
C) Incorrect, Plausible because 10 REM is the Emergency Exposure Limit for individuals protecting valuable property
D) Correct. For a lifesaving activity with the worker fully aware of the risks involved, the limit for dose exposure is >25 REM.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	
Group #	1	
K/A #	2.3.4	
Importance Rating	3.2	

K/A Statement: Knowledge of radiation exposure limits under normal or emergency conditions (CFR 41.12)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
1. RP-AA-203, Exposure Control and Authorization, Rev. 05

Proposed references to be provided to applicants during examination: None

Learning Objective: NGET Radiation Worker

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge ___
Comprehension or Analysis X

Level of difficulty: (1-5) 3

10 CFR Part 55 Content:
55.41 12
55.43 _____

Comments:

Question: 72

Which of the following actions requires the Control Room to notify Radiation Protection that areas in both the Reactor Building and Radwaste may require upgrading of their radiation area postings?

A. Placing a RWCU Filter Demineralizer in service

B. Operation of the Traversing In- Core Probe system

C. Swapping Fuel Pool Cooling Filter Demineralizer “A” to “B”

D. Flushing the Residual Heat Removal System drain lines following restoration from Shutdown Cooling operation.

Answer: D

Explanation:

A) Incorrect – It's plausible that backwashing a Reactor Water Cleanup filter demineralizer may affect radiation levels, however there are no precautions about changes in radiation levels during this procedure.
B) Incorrect – Plausible because operation of the Traversing In-Core Probe system may affect radiation levels in the Reactor Building. However they will NOT affect radiation levels in Radwaste.
C) Incorrect – Plausible because backwashing a Fuel Pool Cooling Filter/Demineralizer requires the operator to notify Radiation Protection prior to and after backwashing. However there are no requirements for reposting of Rad areas when conducting this evolution.
D) Correct – The Control Room must notify Radiation Protection that the flushing the drain lines to remove hot spots will cause radiation levels in the Reactor Building and Radwaste. Radiation Protection must survey the areas before the flush and during the flush to monitor its effectiveness.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	
Group #		
K/A #	G2.3.14	
Importance Rating	3.4	

K/A Statement: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (CFR: 41.12 / 43.4 / 45.10)

Technical Reference(s): (Attach if not previously provided including version/revision number.)

1. LN-1000 Residual Heat Removal System, Rev. 20
2. QCOP-1000-34 RHR/RHRSW System Flush to remove hot spots in drain lines, Rev. 19

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________________ (as available)

Question Source: Bank # X
Modified Bank # _______ (Note changes or attach parent)
New _______

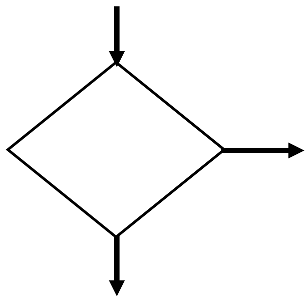
Question History: Last NRC Exam Duane Arnold 2013 #74

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	X
Level of Difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	12
	55.43	______

Comments: Modified answer D to a procedure that is used at Quad Cities.

Question: 73

While reviewing the EOP flowcharts you come across a symbol as is indicated below.



What does this symbol indicate?

A. Decision Step

B. Hold/Wait Point

C. Instructional Step

D. Concurrent Execution

Answer: A

Explanation:

A) Correct.
B) Incorrect. A Hold/Wait Point is depicted by an octagon.
C) Incorrect. An Instruction Step is depicted by a box.
D) Incorrect. Concurrent Execution is depicted by parallel drop down pathways.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	______
	Group #	1	______
	K/A #	2.4.19	
	Importance Rating	3.4	

K/A Statement: Knowledge of EOP layout, symbols, and icons (CFR 41.10)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
1. QCAP 0200-10, Emergency Operating Procedure (QGA) Execution Standards, Rev. 54

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # X
Modified Bank # _______ (Note changes or attach parent)
New _______

Question History: Last NRC Exam DAEC NRC ILT Exam 2015

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

Level of difficulty: (1-5) 2

10 CFR Part 55 Content: 55.41 10
55.43 _____

Comments:

Question: 74

Unit 2 was at full power when a transient occurred resulting in the declaration of an 'ALERT' emergency classification.

The Technical Support Center (TSC) is being staffed, but has not yet achieved minimum staffing.

Which of the following is currently responsible for the "command and control" function during implementation of the emergency plan.

A. Station Emergency Director

B. Shift Manager

C. Unit Supervisor

D. Station Vice President

Answer: B

Explanation:

A) Incorrect. Plausible since the Station Emergency Director will assume "Command and Control" from the Shift Manager once the TSC is activated. However, the TSC is not yet activated.
B) Correct. Since the TSC is not yet activated, the Shift Manager fills this role.
C) Incorrect. Plausible since the Unit Supervisor has "command and control" responsibility for operating the plant to mitigate the emergency.
D) Incorrect. Plausible since the Station Vice President is overall responsible for station operations.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	
Group #		
K/A #	G2.4.37	
Importance Rating	3.0	

K/A Statement: Knowledge of the lines of authority during implementation of the emergency plan. (CFR: 41.10 / 45.13)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
1. EP-QC-1000 Quad Cities Nuclear Power Station Radiological Emergency Plan, Rev. 0

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (as available)

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ______

Level of Difficulty: (1-5) 2

10 CFR Part 55 Content:
55.41 10
55.43 ______

Comments:

Question: 75

Unit 1 Main Generator is currently operating with the following parameters:

- 60 psig hydrogen
- 900 MWe
- 18 KV Generator Terminal voltage
- Generator voltage regulator in Auto

Which of the following MVAR loads will result in a Main Generator limitation being exceeded?

A. 430 MVAR OUT

B. 300 MVAR OUT

C. 25 MVAR IN

D. 100 MVAR IN

Answer: D

Explanation:

A) Incorrect. The generator is being operated at or just below the limit of the 60 psig H2 curve.
B) Incorrect. The generator is being operated just above the limit of the 45 psig H2 curve, but well within the limits of the 60 psig H2 curve.
C) Incorrect. The generator is being operated approx. 25 MVAR above the UEL curve for the given terminal voltage, which is above the curve limit and in accordance with the administrative limit of QCOP 6000-02
D) Correct. The value is below the UEL curve for the given terminal voltage. The UEL should prevent operation at this value.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	____
Group #	1	____
K/A #	2.4.47	
Importance Rating	4.2	

K/A Statement: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material (CFR 41.10)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QCOP 6000-02, Adjusting VARS on the Main Generator, Rev. 22

Proposed references to be provided to applicants during examination:

QCOP 6000-02, ATTACHMENT A UNIT 1(2) GENERATOR CAPABILITY CURVES AUTOMATIC OPERATION (Page 1 ONLY without the NOTES)

Learning Objective: SR-6000A-K36 Given the Main Generator Capability Curve and Generator operating parameters, EXPLAIN why there are limits and DETERMINE if the Main Generator is within the limits of the Main Generator Capability Curve.

Question Source:

Bank #	____	
Modified Bank #	____	(Note changes or attach parent)
New	X	

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	___
	Comprehension or Analysis	X
Level of difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	10
	55.43	___

Comments:

Question: 76

1000 Unit 1 and 2 Loss of Offsite Power

1001 Annunciators in alarm for Unit 1 and Unit 2

- A-4 DIESEL GEN 1/2 TROUBLE
- C-4 DIESEL GEN 1/2 FAIL TO START

1011 Annunciators in alarm for Unit 2 (ONLY)

- A-7 DIESEL GEN 2 TROUBLE
- G-8 DIESEL GENERATOR 2 RELAY TRIP
- H-4 VOLTAGE DEGRADED ON BUS 24-1

1012 QCOA 6100-04 STATION BLACKOUT (SBO) is being implemented on Unit 2.

Which of the following identifies steps of QCOA 6100-04 that are required to be performed?

A. D.3, D.4, D.5, D.7

B. D.2, D4, D.5, D.6

C. D.2, D.5, D.7, D.8

D. D.3, D.5, D.6, D.8

Answer: B

Explanation:
After the loss of offsite power and failure of the Unit 1/2 EDG, Bus 14-1 is energized from the Unit 1 EDG and Bus 24-1 is powered from the Unit 2 EDG. When the Unit 2 EDG fails at 1011, Unit 2 enters a SBO condition. (Bolded steps are required)

D.1 Applicability of SBO procedure confirmed
D.2 Power is available from the opposite Unit for the battery chargers.
D.3 Power is, or will be shortly, available to battery chargers; load shedding NOT required.
D.4 SBO DG 2 required to power Unit 2
D.5 Load shedding/stripping required to prevent overload of emergency supplies.
D.6 RPV Level and Pressure control with HPCI and RCIC required.
D.7 There is no indication that offsite power is, or will be available.
D.8 Opposite unit affected by LOOP but is not in a SBO

A) Incorrect. There is no indication that DC load shedding is required and no indication that offsite power will be available. Plausible as DC load shedding would be required if power were not available from the opposite unit, and step D.7 would be appropriate action if offsite power were to be restored quickly.
B) Correct. See above step descriptions.
C) Incorrect. There is no indication that offsite power will be available and although Unit 1 was also affected by the LOOP, it has not experienced a SBO; therefore Unit 1's SBO EDG should not be powering bus 14-1. Plausible if offsite power were quickly restored.
D) Incorrect. There is no indication that DC load shedding is required and no indication that offsite power will be available and although Unit 1 was also affected by the LOOP, it has not experienced a SBO; therefore Unit 1's SBO EDG should not be powering bus 14-1. Plausible as DC load shedding would be required if power were not available from the opposite unit and if Unit 1 SBO EDG were supplying that powe.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	295003AA2.01	
Importance Rating		3.7

K/A Statement: Partial or Complete Loss of A.C. Power: Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Cause of partial or complete loss of A.C. power (CFR: 41.10 / 43.5 / 45.13)

Technical Reference(s):
(Attach if not previously provided, including version/revision number.)

1. QCGP 2-3 Reactor Scram, Rev. 87
2. QCOA 6100-04, Station Blackout, Rev. 23
3. QCOA 6100-03 Loss of Offsite Power, Rev. 42
4. QCAN 902-8 VOLTAGE DEGRADED ON BUS 24-1, Rev. 0
5. QCAN 901(2)-8 A-7 DIESEL GEN 2 TROUBLE, Rev. 5
6. QCAN 901(2)-8 G-8 DIESEL GENERATOR 2 RELAY TRIP, Rev. 6

Proposed references to be provided to applicants during examination:

QCOA 6100-04, Station Blackout, Rev. 23

Learning Objective:	S-SBO-K12 Given QCOA 6100-04, PREDICT the intermediate and final plant/system conditions and parameter responses as the procedure is accomplished.
Question Source:	Bank # ____ Modified Bank # ____ (Note changes or attach parent) New __X__
Question History:	Last NRC Exam __N/A__
Question Cognitive Level:	Memory or Fundamental Knowledge ____ Comprehension or Analysis __X__
Level of Difficulty:	(1-5) __3__
10 CFR Part 55 Content:	55.41 ____ 55.43 _B.5_

Comments:

SRO Level Justification: Assessment of Facility Conditions and Selection of appropriate procedures during normal, abnormal, and emergency situations.

Question: 77

Unit 2 is operating at 30% power when a pressure transient occurs:

- Reactor pressure lowers to 780 psig (lowest reached)
- Reactor water level rises to +50 inches (highest reached)

Subsequently, operators stabilize reactor pressure and water level in their normal bands.

Which of the following identifies:

(1) the current plant status;

AND

(2) 10 CFR 50.72 reporting requirement?

A. (1) reactor automatic SCRAM
(2) 1 hour ENS report

B. (1) reactor automatic SCRAM
(2) 4 hour ENS report

C. (1) reactor operating at approximately 30% RTP on turbine bypass valves
(2) 4 hour ENS report

D. (1) reactor operating at approximately 30% RTP on turbine bypass valves
(2) no ENS report required

Answer: B

Explanation:

A) Incorrect. The first part is correct. The second is plausible if the applicant confuses the plant conditions and thinks a Safety Limit violation has occurred (SL is reactor pressure < 685 psig vs. 780 psig in stem). A Safety Limit violation requires the NRC to be notified within one hour.
B) Correct. An MSIV closure (Group I Isolation) on low reactor pressure with the Mode switch in RUN will occur based on pressure down to 785 psig, therefore resulting in an Automatic SCRAM on MSIVs less than 90% open. An actuation of the RPS with the reactor critical requires a 4 hour ENS report IAW 10 CFR 50.72 and LS-AA-1020.
C) Incorrect. Plausible if the examinee does not recognize that RPV pressure has dropped below the MSIV closure set-point and focuses only on high reactor water level trip of the turbine with power less than 38.5% (within the capacity of the turbine bypass valves).
D) Incorrect. Plausible if the examinee does not recognize that RPV pressure has dropped below the MSIV closure set-point and focuses only on high reactor water level trip of the turbine with power less than 38.5% (within the capacity of the turbine bypass valves). If there was only a turbine trip, the examinee would correctly assess that an ENS notification would not be required per LS-AA-1020.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	____	1
Group #	____	1
K/A #	295006AA2.04	
Importance Rating	4.1	

K/A Statement: SCRAM: Ability to determine and/or interpret the following as they apply to SCRAM: Reactor pressure (CFR 43.5)

Technical Reference(s):
(Attach if not previously provided, including version/revision number.)

1. LS-AA-1020, Reportability Tables and Decision Trees, Rev. 26
2. QCGP 2-3, Reactor SCRAM, Rev.
3. Technical Specification Safety Limit 2.1.1.1
4. Technical Specification 3.3.6.1

Proposed references to be provided to applicants during examination:

LS-AA-1020, Reportability Tables and Decision Trees, Rev. 26

Learning Objective: S-REPT-K03 and SR-1603-K10

Question Source:
Bank # __X__
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam Hatch NRC ILT Exam 2012

Question Cognitive Level:	Memory or Fundamental Knowledge	___
	Comprehension or Analysis	_X_
Level of difficulty:	(1-5) __2__	
10 CFR Part 55 Content:	55.41 ____	
	55.43 1 and 5	

Comments:
Changes made to two distractors and a Quad Cities procedure reference was incorporated in the stem.

SRO only because of link to 10CFR55.43(b)(1): Conditions and limitations in the facility license. (Reporting Requirements)

Question: 78

Unit 1 is refueling.

- The bridge is over the core with a fuel bundle loaded on the grapple.
- The grapple is in the full up position.
- The bundle is being moved from the core to the spent fuel pool.

Then, the Refuel SRO on the bridge reports a lowering pool level.

The fuel bundle was placed in an emergency set-down location and the refuel floor was evacuated.

The lowering pool level was stopped by shutting the Shutdown Cooling Suction isolation valves.

Spent Fuel Pool Level (LI-1-1901-121) stabilized at approximately 18'.

Additional indications received as pool level lowered include:

- Annunciator 901-4 B-24 Fuel Pool Storage Hi/Lo Level has alarmed

- Annunciator 901-3 B-1 Refuel Floor Hi Radiation has alarmed
- Annunciator 901-3 G-16 Fuel Pool Channel A Hi Radiation has alarmed
- Annunciator 901-3 H-16 Fuel Pool Channel B Hi Radiation has alarmed

- Both Standby Gas Treatment trains auto start
- Both Unit 1 Fuel Pool Cooling Pumps have tripped.

What is the required Emergency Classification for this event?

A. UNUSUAL EVENT

B. ALERT

C. SITE AREA EMERGENCY

D. GENERAL EMERGENCY

Answer: A

Explanation:

A) Correct. An UNUSUAL EVENT due to 1) UNPLANNED water level drop in the REFUELING PATHWAY as reported by the Refuel SRO; AND 2) UNPLANNED Area Radiation Monitor reading rise on Fuel Pool Area Radiation Monitors (901-3 G16 and H16)incorrect because of the IMMINENT loss of level above irradiated fuel regardless of mitigation or corrective actions. The high radiation alarms indicate a level much lower than -3 inches (where the Fuel Pool Storage Hi/Lo Level alarm actuates).
B) Incorrect. Declaring an ALERT is incorrect since radiation level do not indicate uncovering of or damage to irradiated fuel (bundle returned to safe location) and spent fuel pool level is well above the EAL threshold.
C) Incorrect. Spent Fuel Pool level has been continuously been maintained well above the EAL threshold.
D) Incorrect. Spent Fuel Pool level has been continuously been maintained well above the EAL threshold.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	295023AA2.05	
Importance Rating		4.6

K/A Statement: Refueling Accidents: Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: Entry conditions of emergency plan (CFR: 41.10/43.5/45.13)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QCAN 901-3 G-16 Refuel Floor Radiation Monitor Channel A Hi Radiation, Rev. 8
2. QCOA 1900-01 Loss of Water Level in the Fuel Storage Pool or Reactor Cavity, Rev. 31
3. EP-AA-1006 Addendum 3 Emergency Action Levels for Quad Cities Station, Rev. 4
4. QCAN 901-3 H-16 Refuel Floor Radiation Monitor Channel B Hi Radiation, Rev. 8
5. QCAN 901-4 B-24 Fuel Storage Pool Water Level High or Low, Rev. 11

Proposed references to be provided to applicants during examination:

EP-AA-1006 Addendum 3 Emergency Action Levels for Quad Cities Station, Rev. 4

Learning Objective: S-0801-K70 Given Refueling Floor Key Parameters 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 EPAA-111 and EP-AA-1006.

Question Source:	Bank #	________
	Modified Bank #	________ (Note changes or attach parent)
	New	X
Question History:	Last NRC Exam	Pilgrim 2013 exam #76
Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	X
Level of Difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	______
	55.43	B.7

Comments:

Pilgrim 2013 exam #76 was used as the base for this question, but was changed to match the K/A.

SRO Level justification: 10 CFR 55.43(b)(7) Fuel Handling Facilities and Procedures to include emergency classifications.

Question: 79

With regards to Technical Specification LCO 3.4.3, Safety and Relief Valves;

_____(1)_____ safety valves must be OPERABLE in order to avoid exceeding the Reactor Coolant System Pressure Safety Limit following a _____(2)_____.

A. (1) Five
(2) turbine trip with failure to SCRAM on turbine stop valve position and no bypass or relief valve operation.

B. (1) Nine
(2) turbine trip with failure to SCRAM on turbine stop valve position and no bypass or relief valve operation.

C. (1) Five
(2) MSIV closure with failure to SCRAM on MSIV position and no relief valve operation.

D. (1) Nine
(2) MSIV closure with failure to SCRAM on MSIV position and no relief valve operation.

Answer: D

Explanation:

A) Incorrect. . (1) Plausible since this is the number of Relief valves that must be operable per TS 3.4.3. (2) Plausible as this is an analyzed overpressure event, but it is not the limiting event.
B) Incorrect. (1) Nine safety valves must be operable. (2) Plausible as this is an analyzed overpressure event, it is not the limiting event though and second statement is correct.
C) Incorrect. (1) Plausible since this is the number of Relief valves that must be operable per TS 3.4.3. (2) MSIV closure with primary SCRAM function and relief valve failure is the limiting overpressure event analyzed.
D) Correct. (1) Nine safety valves must be operable. (2) MSIV closure with primary SCRAM function and relief valve failure is the limiting overpressure event analyzed. The Safety Limit is based on maintaining pressure at the area of the RCS (lowest elevation) experiencing the highest pressure < than 110% of the ASME design value.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	295025 2.2.25	
Importance Rating	4.2	

K/A Statement: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits (CFR 43.2)

Technical Reference(s):
(Attach if not previously provided, including version/revision number.)

1. Technical Specification Bases 3.4.3, Safety and Relief Valves, Rev. 0
2. UFSAR Section 5.2.2.2.3, Safety Valve Steam Flow Capacity, Rev. 12
3. Technical Specification Bases 2.1.2, Reactor Coolant System Pressure Safety Limit, Rev. 31

Proposed references to be provided to applicants during examination: None

Learning Objective: ITS.INTRO.8

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	_X_
	Comprehension or Analysis	____
Level of difficulty:	(1-5) __2__	
10 CFR Part 55 Content:	55.41 ______	
	55.43 __2__	

Comments:

SRO Only Justification: Question requires knowledge of TS Bases that is required to analyze TS required actions and terminology

Question: 80

Unit 2 was at full power when an earthquake exceeding the Operating Basis Earthquake (OBE) occurred.

1	1000	QGA 200, PRIMARY CONTAINMENT CONTROL, is entered due to lowering Torus water level.
2	1005	A plant shutdown in initiated due to the earthquake exceeding the OBE and inability to maintain Torus Water Level within the required band.
3	1015	Shift Manager declares an ALERT emergency classification due to the earthquake that caused damage to the Unit 2 torus.
4	1100	Exelon generates a news release regarding the status of Quad Cities following the earthquake.

Which of the above conditions will require the earliest notification to the NRC Operations Center?

A. 1

B. 2

C. 3

D. 4

Answer: C

Explanation:

A) Incorrect. QGA entry in and of itself does not require NRC notification.
B) Incorrect. This is plausible because a 4 hour report for an event involving a shutdown required by technical specifications is required per 10 CFR 50.72(b)(2)(i). The crew initiated the shutdown based on the inability to maintain suppression pool water level within limits (TS 3.6.2.2) and due to the earthquake exceeding an OBE.
C) Correct. The HOO must be notified of an ALERT declaration within 1 hour of the declaration per 10 CFR 50.72(a)(3).
D) Incorrect. This is plausible because a 4 hour report is required for a news release regarding an event related to the health and safety of the public per 10 CFR 50.72(b)(2)(xi).

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	295030G2.4.30	
Importance Rating		4.1

K/A Statement: Low Suppression Pool Water Level: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator. (CFR: 41.10 / 43.5 / 45.11)

Technical Reference(s):
(Attach if not previously provided, including version/revision number.)

1. QCOA 0010-09 Earthquakes, Rev. 17
2. QGA-200 Primary Containment Control, Rev. 11
3. EP-AA-1006 Addendum 3 Emergency Action Levels for Quad Cities Station, Rev. 4
4. EP-QC-1000 Quad Cities Nuclear Power Station Radiological Emergency Plan, Rev. 0
5. LS-AA-1020, Reportability Tables and Decision Trees, Rev. 26

Proposed references to be provided to applicants during examination:

LS-AA-1020, Reportability Tables and Decision Trees, Rev. 26

Learning Objective: S-REPT-K03 Given an event, DEMONSTRATE the ability to use the Reportability Manual Decision Trees, ESF Actuation Flow Charts, Reporting Requirement Guidance References and/or the electronic reportability manual to locate reportability guidance applicable to the event.

Question Source:
Bank # ________
Modified Bank # ________ (Note changes or attach parent)
New ___X___

Question History: Last NRC Exam __N/A______

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	__X___
Level of Difficulty:	(1-5) __3__	
10 CFR Part 55 Content:	55.41 _____	
	55.43 __B.1__	

Comments:

SRO Level Justification: Conditions and Limitations in the facility license. This question pertains to reporting requirements.

Question: 81

Unit 2 was operating at 100% power, when a large loss of coolant accident occurred (at 0900).

During the course of the accident, Main Chimney Gas Activity Monitor Recorder 1/2-1740-202 indicates as follows:

Time	MN CHIMNEY GAS ACTIVITY	ALARMS
0900	5.0E-7 µCi/cc (low range)	
0915	6.0E-5 µCi/cc (low range)	912-1, F-3, STACK GAS RAD HI
0925	1.2E-4 µCi/cc (low range)	912-1, E-3, STACK GAS RAD HI HI
0945	2.0E-3 µCi/cc (low range)	912-1, E-9, RAD MON SYS A HIGH SCALE
1000	2.7E-3 µCi/cc (low range)	
1045	2.4E0 µCi/cc (mid range)	
1100	2.3E1 µCi/cc (high range)	
1110	2.5E1 µCi/cc (high range)	
1115	2.0E1 µCi/cc (high range)	

Stack flow = 350E3 CFM

Release Rate (µCi/sec) = Chimney Gas Activity (µCi/cc) X 472 X Stack Flow (CFM)

What is the required Emergency Classification for this event?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: C

Explanation:

A) Incorrect. The release rate exceeds 4.38E5 µCi/sec for more than 60 minutes, but it also exceeds the Alert and SAE thresholds as well.
B) Incorrect. The release rate exceeds 3.84E7 µCi/sec for greater than 15 minutes, but it also exceeds the SAE threshold as well.
C) Correct. Release rate exceeds 3.84E8 µCi/sec for greater than 15 minutes.
D) Incorrect. The release rate exceeds 3.8E9 µCi/sec but does not remain so for greater than 15 minutes per the data provided.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	____	1
Group #	____	1
K/A #	295038EA2.02	
Importance Rating	3.3	

K/A Statement: HIGH OFF-SITE RELEASE RATE: Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Total number of curies released (CFR 43.5)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. EP-AA-1006, Addendum 3, Emergency Action Levels for Quad Cities Station, Rev. 4
2. CY-QC-120-735, Main Chimney and Reactor Vent Noble Gate Release Rate, Rev. 9
3. EP-AA-110-200, Dose Assessment, Rev. 8
4. LIC-1702, Chimney Radiation Monitoring
5. QCOA 1700-01, Abnormal Main Chimney Radiation, Rev. 10
6. QCOA 1700-02, High Radiation Detected on Eberline Radiation Monitoring System, Rev.10

Proposed references to be provided to applicants during examination:

EP-AA-1006, Addendum 3, Emergency Action Levels for Quad Cities Station, Rev. 4

Learning Objective: SR-1702-K20

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	___
	Comprehension or Analysis	_X_
Level of difficulty:	(1-5) __3__	
10 CFR Part 55 Content:	55.41 ____	
	55.43 _5__	

Comments:

SRO Justification: Analysis and interpretation of a radioactive release, including comparison to emergency plan criteria.

Question: 82

Unit 1 is operating at 100% power when a fire breaks out in the HPCI room.

- 901-3 F-12 HPCI PUMP AREA HI TEMP is in alarm.
- 901-3 D-12 HPCI PUMP LOW FLOW is in alarm.
- MO 1-2301-3, HPCI TURB STM SPLY VLV is OPEN.
- HPCI Pump Discharge Pressure is approximately 0 psig.
- HPCI Pump Flow is 0 gpm.
- 1/2A and 1/2B Fire Pumps are running
- FIRE PROT SYSTEM ALARM at Panel 912-1 is in alarm
  [FAS Device 81-14, UNIT 1 HPCI VAULT FIRE]

In addition to QCOA 0010-12 FIRE/EXPLOSION, which of the following procedures must be used to mitigate the event?

(1) QGA 300, SECONDARY CONTAINMENT CONTROL
(2) QCGP 2-1, NORMAL UNIT SHUTDOWN
(3) QCGP 2-3, REACTOR SCRAM
(4) QCOA 2300-01, HPCI AUTOMATIC INITIATION
(5) QCAN 901-3 D-12, HPCI PUMP LOW FLOW

A. QGA 300 ONLY

B. QGA 300 and QCGP 2-1

C. QGA 300 and QCGP 2-3,

D. QCOA 2300-01 and QCAN 901-3 D-12

Answer: A

Explanation:

A) Correct. The HPCI Room high temperature annunciator is an indication of an entry criteria into QGA 300.
B) Incorrect. The HPCI room is the only Secondary Containment area with a high temperature. Plausible because a normal unit shutdown which would be required per QGA-300 for 2 areas exceeding the maximum safe temperature.
C) Incorrect. The HPCI room is the only Secondary Containment area with a high temperature. Plausible because the fire may cause the HPCI room temperature to exceed the Max Sate value and a Reactor SCRAM would be required per QGA-300 for any areas exceeding the maximum safe temperature, but only if a Primary System is discharging into that area.
D) Incorrect. QGA 300 entry is required. Plausible because there was a spurious opening of HPCI steam admission valve, which would be one of the indications of an automatic HPCI initiation, and there is no flow on HPCI, but this is due to the spurious opening of the HPCI steam admission valve without injection.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	600000G2.4.45	
Importance Rating		4.3

K/A Statement: Plant Fire On Site: Ability to prioritize and interpret the significance of each annunciator or alarm (CFR: 41.10 / 43.5 / 45.3 / 45.12)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. OP-QC-201-012-1001 Quad Cities On-Line Fire Risk Management Rev. 6
2. QCOA 0010-12 FIRE/EXPLOSION, Rev. 47
3. QGA-300 Secondary Containment Control, Rev. 13
4. QCAN 901-3 D-12, HPCI Pump Low Flow, Rev. 7
5. QCOA 2300-01, HPCI Automatic Initiation; Rev 24

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-4101-K03; Given a fire location and the Quad Cities Fire Pre-Plans, DETERMINE which fire pre-plan is appropriate.

Question Source:
Bank # ________
Modified Bank # ________ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	X
Level of Difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	______
	55.43	B.5

Comments:

Scenario is a fire in the HPCI Room that causes the spurious opening of the HPCI steam admission valve.

SRO Level Justification: Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations.

Does the question require one or more of the following:

Yes – assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or a section of a procedure to mitigate or recover, or with which to proceed.

Question: 83

Unit 1 was at 100% power, when a failure in the DEHC system results in reactor pressure rising and approaching the automatic SCRAM setpoint.

A manual reactor SCRAM was attempted using the Manual SCRAM pushbuttons.

- Reactor Mode Switch is in SHUTDOWN.
- RPS Lights are NOT lit.
- ARI was initiated.
- Reactor Recirculation pumps were tripped.
- 901-5, A-1, SCRAM VALVE AIR SUPPLY LOW PRESSURE is in alarm
- APRM Downscale lights energized after RPV water level was intentionally lowered to -100 inches
- Torus Temperature stable at 140°F

Select the answer that specifies:

(1) The current RPV level control strategy.

AND

(2) The reason for limiting the injection rate.

A. (1) Promptly reestablish injection and maintain RPV water level between -162” and -100”
(2) Prevent core damage resulting from rapid uncontrolled heating of the fuel.

B. (1) Continue to lower RPV level to TAF, then maintain RPV water level between -162” and -142”
(2) Prevent core damage resulting from rapid uncontrolled heating of the fuel.

C. (1) Continue to lower RPV level to TAF, then maintain RPV water level between -162” and -142”
(2) Prevent core damage resulting from rapid uncontrolled cooling of the fuel cladding.

D. (1) Promptly reestablish injection and maintain RPV water level between -162” and -100”
(2) Prevent core damage resulting from rapid uncontrolled cooling of the fuel cladding.

Answer: A

Explanation:

A) Correct. (1) With reactor power <5% (APRM Downscale lights lit, injection is to be reestablished and maintained in the specified band. (2) rapidly increasing injection may cause a large net increase in positive reactivity due to increased subcooling at the core inlet, reduction of the core void fraction, and, if boron has been injected, the removal of boron from the core region. The subsequent power excursion may be large enough to cause substantial damage to the core and the RPV.
B) Incorrect. (1) Plausible as this would be the strategy if power remained above 5%. (2) Correct reason; See 'A"
C) Incorrect. (1) Plausible as this would be the strategy if power remained above 5%. (2) Plausible since colder water is being injected, but the rapid heating of the fuel will be much greater and net effect will be a heat-up of the cladding.
D) Incorrect. (1) With reactor power <5% (APRM Downscale lights lit, injection is to be reestablished and maintained in the specified band. (2) Plausible since colder water is being injected, but the rapid heating of the fuel will be much greater and net effect will be a heat-up of the cladding.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	____	1
Group #	____	2
K/A #	295007 2.4.20	
Importance Rating	4.3	

K/A Statement: HIGH REACTOR PRESSURE: Knowledge of the operational implications of EOP warnings, cautions, and notes (CFR 41.10 / 43.5 / 45.13)

Technical Reference(s):
(Attach if not previously provided, including version/revision number.)

1. QGA 101, RPV Control (ATWS), Rev. 15
2. QCOA 0201-03, Reactor High Pressure, Rev. 31
3. QCGP 2-3, Reactor SCRAM, Rev. 87
4. LIC-0703, LPRM/APRM
5. L-QGA101, QGA 101, RPV Control (ATWS)

Proposed references to be provided to applicants during examination: None

Learning Objective: S-0001-K70b and SR-0001-K060

Question Source:
Bank # _______
Modified Bank # ___ (Parent attached)
New X

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	___
	Comprehension or Analysis	_X_
Level of difficulty:	(1-5)	__3__
10 CFR Part 55 Content:	55.41	____
	55.43	__5__

Comments:

SRO Justificaiton: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Question: 84

Unit 1 is at 100% power.

1000 RBCCW Return from Drywell Outboard Isolation Valve 1-3703 has a dual indication.

1002 Annunciator in alarm:

- 901-4 G-3 RECIRCULATION PUMP A SEAL COOLING WATER LOW FLOW
- 901-4 G-7 RECIRCULATION PUMP B SEAL COOLING WATER LOW FLOW

1005 Recirc Pumps 1A and 1B Seal Cooling Water temperatures is 105°F and rising at 1 ˚F per minute.

1020 The following conditions are observed:

- Recirc Pumps 1A and 1B Seal Cooling Water temperature alarms received; temperatures120˚F and continuing to rise at 1˚F per minute.
- Drywell pressure 1.0 psig and rising at 0.02 psig per minute.
- Drywell average air temperature 125˚F and rising at 2 ˚F per minute.

Which of the following actions is required?

A. Reduce Recirc Pump Speed in an attempt to reduce Recirc Pump Seal Cooling Water Temperatures.

B. Initiate a normal plant shutdown to comply with Tech Spec LCO action statements.

C. Perform an Emergency Power Reduction to reduce heat input into the Drywell.

D. SCRAM the reactor and trip both Recirc Pumps within one minute due to the loss of RBCCW cooling.

Answer: A

Explanation:

A) Correct. The QCAN for low Recirc pump seal cooling water flow directs a reduction in Recirc pump speed in an attempt to reduce temperatures.
B) Incorrect. At the current rate the TS LCO for Drywell Temperature will not be exceeded for 12.5 minutes and a shutdown would not be required for at least 8 hours. The Drywell pressure LCO will not be exceeded for 25 minutes and then 1 hour is permitted to restore pressure below the limit. A SCRAM may be required if pressure continues to rise, but is not expected to be initiated for approx. an additional 25 minutes.
C) Incorrect. While an Emergency Power reduction may be warranted if cooling cannot be restored, it is not called for at this time.
D) Incorrect. A total loss of RBCCW cooling to the Recirc pumps has not occurred and a SCRAM is not justified at this time.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #	295020AA2.03	
Importance Rating		3.7

K/A Statement: Inadvertent Containment Isolation: Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION: Reactor power (CFR: 41.10 / 43.5 / 45.13)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. LN-3700 Reactor Building Closed Cooling Water, Rev. 3
2. LN-0202 Reactor Recirculation System, Rev. 5
3. QCAN 901(2)-4 G-3 Recirculation Pump A Seal Cooling Water Low Flow, Rev. 10
4. QCAN 901(2)-4 G-7 Recirculation Pump B Seal Cooling Water Low Flow, Rev. 7
5. TS LCO 3.6.1.4, 3.6.1.5 and Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-1601-K29 Given Containment Systems key parameter indications and various plant conditions, DETERMINE, from memory, if the Containment Systems Tech Spec LCOs have been met.

SRN-0202-K23, Given a Reactor Recirculation System operating mode and various plant conditions, PREDICT how the Reactor Recirculation System will be impacted by the following support system failures:

Question Source:

Bank #	________
Modified Bank #	________ (Note changes or attach parent)
New	X

Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
Level of Difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	
	55.43	B.1

Comments:

SRO Level justification: Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations.

Question: 85

A large seismic event has occurred resulting in a loss of offsite power to both units.

Unit 1 experienced a small break loss of coolant accident

- QGA 100 RPV CONTROL is being implemented
- QGA 200 PRIMARY CONTAINMENT CONTROL is being implemented
- HPCI is being utilized to maintain RPV level

Subsequently, a steam leak developed in the HPCI room.

The following plant conditions presently exist on Unit 1:

Reactor	Primary Containment	Secondary Containment
All rods in	DW Pressure: 2.0 psig	RB Exhaust Vent Rad Monitor is in alarm; reading: 5 mr/hr
RPV level: +8 inches	DW Temperature: 240ºF	HPCI Room Rad Monitor is in alarm; reading 120 mr/hr
RPV pressure: 900 psig	Torus Level: +2 inches	HPCI Room Temperature: 105ºF
	Torus Temperature: 100ºF	

(1) Operation of the Standby Gas Treatment System is…

(2) Which action should the Unit Supervisor order with regards to the HPCI system?

A. (1) required
(2) Shutdown and isolate HPCI

B. (1) NOT required
(2) Shutdown and Isolate HPCI

C. (1) required
(2) Continue to maintain RPV level with HPCI

D. (1) NOT required
(2) Continue to maintain RPV level with HPCI

Answer: A

Explanation:

A) Correct. (1) QGA 300 (global override) requires verification of RB Vent isolation and SBGT system start if RB Exhaust Ventilation radiation is > 3 mr/hr; 2) HPCI room radiation is above 100 mr/hr (a QGA 300 entry condition), and with other systems available to maintain RPV level, HPCI is required to be isolated.
B) Incorrect. (1) Plausible since radiation level is below the analytical limit (10 mr/hr) for RB Vent Isolation and SBGT start. (2) Plausible as HPCI room radiation above 100 mr/hr (a QGA 300 entry condition), and HPCI should be shutdown and isolated.
C) Incorrect. (1) Plausible since QGA 300 (global override) requires verification of RB Vent isolation and SBGT system start if RB Exhaust Ventilation radiation is > 3 mr/hr; (2) Plausible since HPCI would likely be a preferred injection source, but HPCI room radiation above 100 mr/hr (a QGA 300 entry condition), and HPCI should be isolated when other systems are available to maintain RPV level.
D) Incorrect. . (1) Plausible since radiation level is below the analytical limit (10 mr/hr) for RB Vent Isolation and SBGT start. (2) Plausible since HPCI would likely be a preferred injection source, but HPCI room radiation above 100 mr/hr (a QGA 300 entry condition), and HPCI should be isolated when other systems are available to maintain RPV level.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #	295033 2.4.2	
Importance Rating	4.6	

K/A Statement: HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Knowledge of system set points, interlock and automatic actions associated with EOP entry conditions. (CFR 43.5)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QGA 300, Secondary Containment Control, Rev. 13
2. L-QGA300, QGA 300, Secondary Containment Control
3. LIC-7500, Standby Gas Treatment System

Proposed references to be provided to applicants during examination: None

Learning Objective: S-0001-K30 and SR-7500-K07

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	___
	Comprehension or Analysis	_X_
Level of difficulty:	(1-5) __2__	
10 CFR Part 55 Content:	55.41 ____	
	55.43 _5__	

Comments: K/A did not have a direct SRO Written Exam CFR link, but question is related to CFR 43.4, radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

SRO Only Justification: The question requires the applicant to assess plant conditions and to know the content of procedures in order to select a required course of action.

Question: 86

A drain down to install the reactor vessel head at the end of a refueling outage on Unit 1 is scheduled to begin:

- The Mode Switch is in Shutdown
- Average RCS Temperature is 130˚F
- Replacement of the motors for RHR pumps 1-1002A and 1-1002B is in progress.

The drain down…

A. CAN begin because both pumps in one RHR loop are operable.

B. CANNOT begin because NO RHR shutdown cooling subsystems are operable.

C. CAN begin if the shift verifies an alternate means of decay heat removal is available while monitoring RCS temperature.

D. CANNOT begin unless a risk assessment and risk management actions are taken due to the inoperable RHR shutdown cooling subsystem.

Answer: A

Explanation:

A) Correct. Based on the requirements to show operability per TS 3.9.9, only two pumps in one loop is required for two RHR shutdown cooling subsystems to be considered operable.
B) Incorrect. Two subsystems are operable as long there are two operable pumps in one loop, or one operable pump in each in each loop. Plausible because there could be a misconception that there must be 1 operable RHR pump in each loop for that subsystem to be operable.
C) Incorrect. There are two operable subsystems. Plausible if there is a misconception that there must be at least one operable pump in each loop. This action is required if one REQUIRED RHR shutdown cooling subsystem is inoperable.
D) Incorrect. Plausible if there is a misconception that there must be at least one operable pump in each loop. This is the action required by LCO 3.0.4b for a change in an operating mode or specified condition if the LCO is not met. The LCO is still met regardless of the water level above the RPV flange.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	____	2
Group #	____	1
K/A #	205000G2.2.38	
Importance Rating	____	4.5

K/A Statement: Shutdown Cooling System (RHR Shutdown Cooling Mode): Knowledge of conditions and limitations in the facility license (CFR: 41.7 / 41.10 / 43.1 / 45.13)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. TS 3.9.8 RHR- High Water Level and Bases
2. TS 3.9.9 RHR-Low Water Level and Bases
3. LN-1000 Residual Heat Removal System, Rev. 20

Proposed references to be provided to applicants during examination:

Technical Specification 3.9.8 Residual Heat Removal (RHR) – High Water Level and Bases
Technical Specification 3.9.9 Residual Heat Removal (RHR) – Low Water Level and Bases

Learning Objective: S-1000-K27 ANALYZE a given condition that may impact the operability of the RHR or RHRSW systems, (i.e. component/controller failure, Clearance) using P&ID/C&IDs, E-prints and Tech Specs, if necessary, and DETERMINE if the RHR/RHRSW meets Tech Spec operability requirements.

Question Source:
Bank # ________
Modified Bank # ________ (Note changes or attach parent)
New ___X___

Question History: Last NRC Exam __________

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	__X__
Level of Difficulty:	(1-5)	_3___
10 CFR Part 55 Content:	55.41	_____
	55.43	__B.1_

Comments: Question is modelled after question# 80 used on the Columbia 2015 but question scenario is sufficient different to categorize this question as NEW.

SRO Level Justification: Conditions and Limitations in the Facility License.

Question: 87

Unit 2 is at 100% power.

- 0900 on March 26th
  - 902-5, G-6, STANDBY LIQ CONTROL TANK HI/LO TEMP is in alarm.
  - Local SBLC Tank temperature is 87ºF.
  - Tank heater breaker at MCC 29-1 is tripped

- 1300 on March 26th
  - SBLC storage tank boron concentration is 15.0%
  - SBLC storage tank temperature continues to lower and is 77ºF

(1) Reactor coolant temperature must be ≤ 212ºF by….

(2) Per Technical Specification Bases 3.1.7, SBLC is required to remain operable in Mode 3 to ensure….

A. (1) 0500 on March 28th
(2) offsite doses remain within 10CFR50.67 limits following a LOCA

B. (1) 0900 on March 28th
(2) shutdown capability exists for the subsequent plant cooldown

C. (1) 0500 on March 28th
(2) shutdown capability exists for the subsequent plant cooldown

D. (1) 0900 on March 28th
(2) offsite doses remain within 10CFR50.67 limits following a LOCA

Answer: D

Explanation:

A) Incorrect. Plausible as time to reach Mode 4 is 44 hours from the time of entering condition B (both trains of SBLC inoperable), but the starting time for entering the LCO is not actuation of the low temperature alarm, but when the temperature exits the allowable region of Figure 3.1.7-2. The second part of the answer is correct.
B) Incorrect: The first portion is correct. The second part is plausible as the reason for operability in Mode 1 and 2. In Mode 3 with all rods in, the reactor will be shutdown under all conditions, and will not require additional boron for cold shutdown.
C) Incorrect. Plausible as time to reach Mode 4 is 44 hours from the time of entering condition B (both trains of SBLC inoperable), but the starting time for entering the LCO is not actuation of the low temperature alarm, but when the temperature exits the acceptable operating region of Figure 3.1.7-2. The second part is plausible as the reason for operability in Mode 1 and 2. In Mode 3 with all rods in, the reactor will be shutdown under all conditions, and will not require additional boron for cold shutdown.
D) Correct. The allowable time is 44 hours. Mode 4 must be achieved due to the LOCA concern. In MODES 1, 2, and 3, the SLC System must be OPERABLE to ensure that offsite doses remain within 10 CFR 50.67 limits following a LOCA involving significant fission product releases. The SBLC System is designed to maintain suppression pool pH at or above 7 following a LOCA to ensure that iodine will be retained in the suppression pool water.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	____	2
	Group #	____	1
	K/A #	211000 A2.05	
	Importance Rating	3.4	

K/A Statement: STANDBY LIQUID CONTROL SYSTEM: Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of SBLC Tank Heaters (CFR 43.2)

Technical Reference(s):
(Attach if not previously provided, including version/revision number.)

1. Technical Specification Bases B3.1.7, Standby Liquid Control System
2. QCOA 1100-01, SBLC Tank Abnormal Temperature, Rev. 17

Proposed references to be provided to applicants during examination:

TS LCO 3.1.7 Sodium Penta-Borate Temperature Requirements, pages 1-6 (include SR and Figures

Learning Objective: S-1100-K27

Question Source: Bank # X
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam Browns Ferry NRC ILT Exam 2013

Question Cognitive Level:	Memory or Fundamental Knowledge	___
	Comprehension or Analysis	_X_
Level of difficulty:	(1-5) __3__	
10 CFR Part 55 Content:	55.41 ____	
	55.43 __2__	

Comments:
This K/A does not have a specific 55.43 link, but this question does require Technical Specification Bases knowledge and therefore this question can be linked to 43.2.

SRO Only Justification: The question requires knowledge of Technical Specification Bases, which is required to analyze TS-required actions and terminology.

Question: 88

Unit 1 is shutting down for a refuel outage.

- The Reactor Mode Switch is in Startup/Hot Standby.
- ALL Control Rods have been individually inserted to position 00.
- RPV water temperature is 480° F and LOWERING.
- RPV pressure is 600 psig and LOWERING.

QCOP 0500-07, BYPASSING "A" CHANNEL OF THE REACTOR MODE SWITCH TO SHUTDOWN SCRAM, is in progress with the bypass jumpers installed.

(1) Prior to placing the Reactor Mode Switch in SHUTDOWN, what Tech Spec actions (if any) are required?

(2) When the Reactor Mode Switch is subsequently placed in SHUTDOWN, …

A. (1) RESTORE RPS trip capability within 1 hour.
   (2) NO SCRAM signals (1/2 OR FULL) will be initiated.

B. (1) NO Tech Spec Actions are required.
   (2) a 1/2 SCRAM will be initiated.

C. (1) RESTORE RPS trip capability within 1 hour.
   (2) a 1/2 SCRAM will be initiated.

D. (1) NO Tech Spec Actions are required.
   (2) NO SCRAM signals (1/2 OR FULL) will be initiated.

Answer: C

Explanation:

A) Incorrect. (1) This part is incorrect because the jumpers ONLY bypass the "A" channel, the "B" channel will still receive a 1/2 scram. (2) This part is correct. At this Rx temp with the Mode switch still in Startup / Hot Standby the Unit is still in MODE 2. Mode 3 (hot shutdown) is NOT entered until the mode switch is physically moved to SHUTDOWN. Therefore, for the time that the jumpers are installed, while the unit is still in MODE 2, TS 3.3.1.1 Condition C applies because the required FUNCTION 11 (TS Table 3.3.1.1.-1) is NOT maintained. Condition C allows 1 hour to restore RPS trip capability.

B) Incorrect. (1) This part is correct because the jumpers ONLY bypass the "A" channel, the "B" channel will still receive a 1/2 scram. (2) This part is incorrect. At this Rx temp with the Mode switch still in Startup / Hot Standby the Unit is still in MODE 2. Mode 3 (hot shutdown) is NOT entered until the mode switch is physically moved to SHUTDOWN. Therefore, for the time that the jumpers are installed, while the unit is still in MODE 2, TS 3.3.1.1 Condition C applies because the required FUNCTION 11 (TS Table 3.3.1.1.-1) is NOT maintained. Condition C allows 1 hour to restore RPS trip capability.

C) Correct. (1) The jumpers ONLY bypass the "A" channel, the "B" channel will still receive a 1/2 scram. (2) At this Rx temp with the Mode switch still in Startup / Hot Standby the Unit is still in MODE 2. Mode 3 (hot shutdown) is NOT entered until the mode switch is physically moved to SHUTDOWN. Therefore, for the time that the jumpers are installed, while the unit is still in MODE 2, TS 3.3.1.1 Condition C applies because the required FUNCTION 11 (TS Table 3.3.1.1.-1) is NOT maintained. Condition C allows 1 hour to restore RPS trip capability.

D) Incorrect. Neither par is correct. See previous explanations.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	212000A2.16	
Importance Rating		4.1

K/A Statement: Reactor Protection System: Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Changing mode switch position (CFR: 41.5 / 45.6)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. Technical Specification 3.3.1.1 and Basis

Proposed references to be provided to applicants during examination:

Technical Specification 3.3.1.1 and Basis

Learning Objective:	SR-0500-K32 Given Reactor Protection System operability status OR key parameter indications, various plant conditions and a copy of Tech Specs, DETERMINE Tech Spec compliance and required actions, if any.
Question Source:	Bank # ___X___ Modified Bank # ________ (Note changes or attach parent) New ________
Question History:	Last NRC Exam __N/A (Quad Cities Training Bank)___
Question Cognitive Level:	Memory or Fundamental Knowledge _____ Comprehension or Analysis __X__
Level of Difficulty:	(1-5) __3__
10 CFR Part 55 Content:	55.41 _____ 55.43 _B.2__

Comments:

SRO Level Justification: Facility Operating Limitations in the Technical Specifications and their Basis.

Question: 89

Unit 1 was operating at 100% power when a reference leg tap failure occurred on the ‘A’ side Narrow and Medium Range level instruments.

(1) With no operator action, the DFWLCS response __________ result in a full RPS actuation.

(2) To minimize the plant transient from this event, the Unit Supervisor will direct operators to __________.

A. (1) will NOT
   (2) take remote/manual control of the FRVs IAW QCOP 0600-18, “Main Feedwater Regulator Operation” and restore level with the ‘B’ reactor level instruments.

B. (1) will
   (2) block failed ‘A’ level instrument inputs to the DFWLCS IAW QCOP 0600-21, “Operation of the Feedwater Level Control System”.

C. (1) will
   (2) take remote/manual control of the FRVs IAW QCOP 0600-18, “Main Feedwater Regulator Operation” and restore level with the ‘B’ reactor level instruments.

D. (1) will NOT
   (2) block failed ‘A’ level instrument inputs to the DFWLCS IAW QCOP 0600-21, “Operation of the Feedwater Level Control System”.

Answer: C

Explanation:

A) Incorrect. (1) A failure of the ‘B’ side Narrow and Medium Range level instruments reference leg tap would result in the ‘B’ side level instrument input to DWFWLCS being rejected and therefore have minimal impact on actual RPV level. Part (2) is correct.
B) Incorrect. The ‘A’ side Narrow and Medium Range level instruments will fail High (+60”). The ‘B’ side Narrow and Medium Range level instruments will read actual RPV water level, but the ‘B’ side Narrow Range level instrument input to the DFWLCS will be rejected. The failed level instrument input to the DFWLCS will result in a level error which will close the FRVs, causing the actual RPV level to lower, and resulting in a full RPS actuation. Action to block inputs to DFWLCS is a long term action that will not be timely in addressing plant transient.
C) Correct. Failed reference leg tap on ‘A’ instruments affects 2 of 3 level inputs to DFWLCS. As a result the system is “tricked” into rejecting the correct reading ‘B’ NR level instrument. A failed reference leg tap makes level indicate high (+60 inches). This will cause the FRVs to want to close lowering feed flow. When actual level reaches the RPV low level SCRAM setpoint, ‘B’ level instruments will pass a SCRAM signal to RPS. After the SCRAM, the Predefined SCRAM profile will NOT occur as DFWLCS thinks level is > 34 inches and FRVs will continue to close causing level to further decrease. Identifying this casualty in a timely manner and taking FRVs to manual will give the operators a chance to control level and avoid a SCRAM.
D) Incorrect. A failure of the ‘B’ side Narrow and Medium Range level instruments reference leg tap would result in the ‘B’ side level instrument input to DWFWLCS being rejected and therefore have minimal impact on actual RPV level. Action to block inputs to DFWLCS is a long term action that will not be timely in addressing plant transient.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	259002 A2.03	
Importance Rating	3.7	

K/A Statement: REACTOR WATER LEVEL CONTROL SYSTEM: Ability to (a) predict the impacts of the following on the REACTOR WATER LEVEL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of reactor water level input. (CFR 43.5)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QCOP 0600-18, Main Feedwater Regulator Operation, Rev. 25
2. QCOP 0600-21, Operation of the Feedwater Level Control System, Rev. 21
3. LIC-0600, Digital RPV Level Control

Proposed references to be provided to applicants during examination: None

Learning Objective: SR-0600-K26

Question Source:	Bank #	___
	Modified Bank #	_
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	_
	Comprehension or Analysis	X
Level of difficulty:	(1-5)	4
10 CFR Part 55 Content:	55.41	_
	55.43	5

Comments:
SRO Only Justification: Requires the assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure to mitigate or recover, or with which to proceed.

Question: 90

Given the alarm indications on the following page, which of the following identifies procedures required to be implemented?

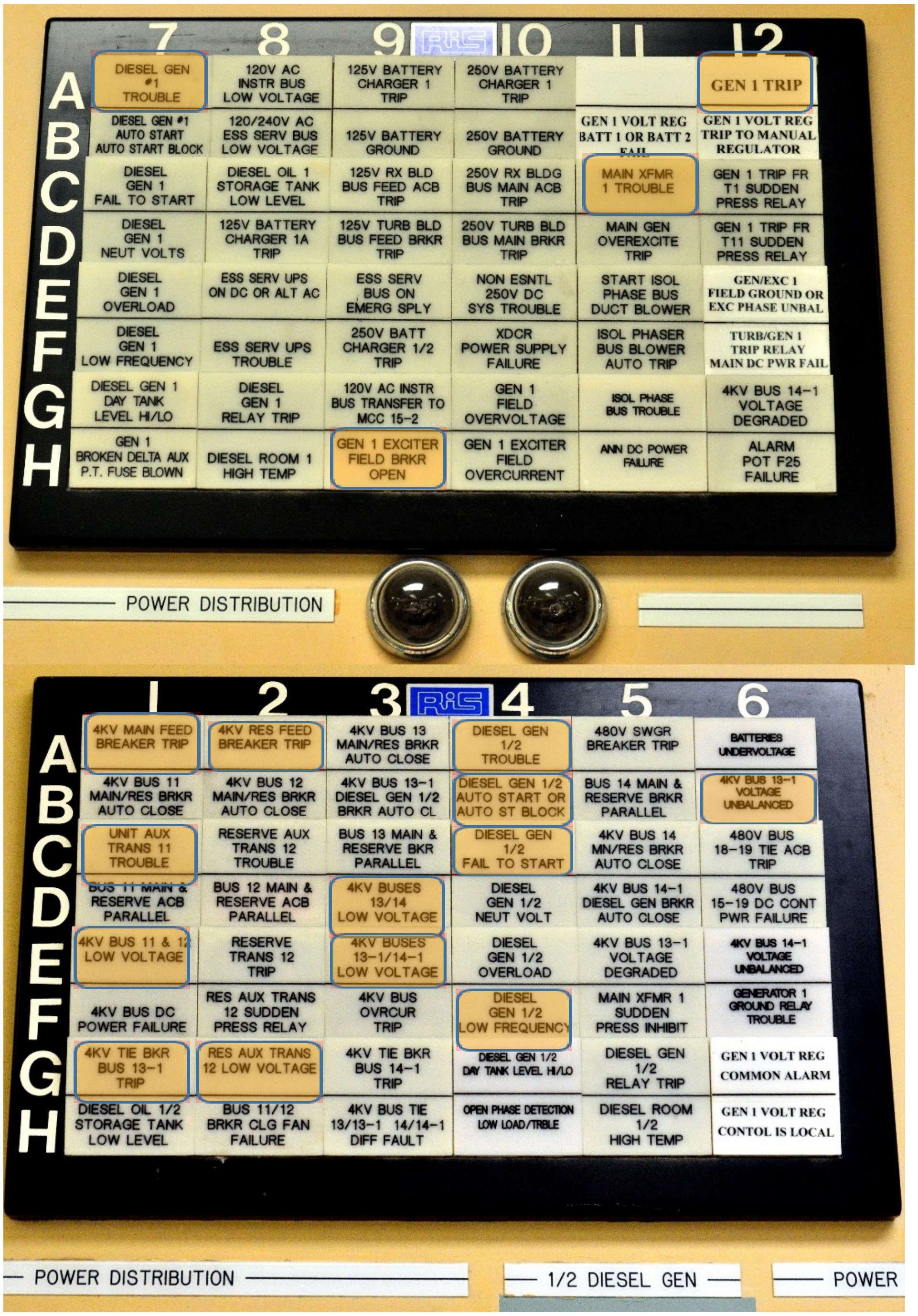
A. QCAN 901-8 A7 DIESEL GEN #1 TROUBLE and
QCOA 6100-03 LOSS OF OFFSITE POWER ONLY

B. QCOA 6600-02 DIESEL GENERATOR 1/2 FAILS TO START and
QCOA 6100-03 LOSS OF OFFSITE POWER ONLY

C. QCAN 901-8 A7 DIESEL GEN #1 TROUBLE,
QCOA 6600-02 DIESEL GENERATOR 1/2 FAILS TO START, and
QCOA 6100-03 LOSS OF OFFSITE POWER

D. QCAN 901-8 A7 DIESEL GEN #1 TROUBLE,
QCOA 6600-02 DIESEL GENERATOR 1/2 FAILS TO START, and
QCOA 6100-04 STATION BLACKOUT

PANEL 901-8


7
8
9
10
11
12
A
B
C
D
E
F
G
H
DIESEL GEN #1 TROUBLE
120V AC INSTR BUS LOW VOLTAGE
125V BATTERY CHARGER 1 TRIP
250V BATTERY CHARGER 1 TRIP
GEN 1 TRIP
DIESEL GEN #1 AUTO START AUTO START BLOCK
120/240V AC ESS SERV BUS LOW VOLTAGE
125V BATTERY GROUND
250V BATTERY GROUND
GEN 1 VOLT REG BATT 1 OR BATT 2 FAIL
GEN 1 VOLT REG TRIP TO MANUAL REGULATOR
DIESEL GEN 1 FAIL TO START
DIESEL OIL 1 STORAGE TANK LOW LEVEL
125V RX BLD BUS FEED ACB TRIP
250V RX BLDG BUS MAIN ACB TRIP
MAIN XFMR 1 TROUBLE
GEN 1 TRIP FR T1 SUDDEN PRESS RELAY
DIESEL GEN 1 NEUT VOLTS
125V BATTERY CHARGER 1A TRIP
125V TURB BLD BUS FEED BRKR TRIP
250V TURB BLD BUS MAIN BRKR TRIP
MAIN GEN OVEREXCITE TRIP
GEN 1 TRIP FR T11 SUDDEN PRESS RELAY
DIESEL GEN 1 OVERLOAD
ESS SERV UPS ON DC OR ALT AC
ESS SERV BUS ON EMERG SPLY
NON ESNTL 250V DC SYS TROUBLE
START ISOL PHASE BUS DUCT BLOWER
GEN/EXC 1 FIELD GROUND OR EXC PHASE UNBAL
DIESEL GEN 1 LOW FREQUENCY
ESS SERV UPS TROUBLE
250V BATT CHARGER 1/2 TRIP
XDCR POWER SUPPLY FAILURE
ISOL PHASER BUS BLOWER AUTO TRIP
TURB/GEN 1 TRIP RELAY MAIN DC PWR FAIL
DIESEL GEN 1 DAY TANK LEVEL HI/LO
DIESEL GEN 1 RELAY TRIP
120V AC INSTR BUS TRANSFER TO MCC 15-2
GEN 1 FIELD OVERVOLTAGE
ISOL PHASE BUS TROUBLE
4KV BUS 14-1 VOLTAGE DEGRADED
GEN 1 BROKEN DELTA AUX P.T. FUSE BLOWN
DIESEL ROOM 1 HIGH TEMP
GEN 1 EXCITER FIELD BRKR OPEN
GEN 1 EXCITER FIELD OVERCURRENT
ANN DC POWER FAILURE
ALARM POT F25 FAILURE
POWER DISTRIBUTION
1
2
3
4
5
6
4KV MAIN FEED BREAKER TRIP
4KV RES FEED BREAKER TRIP
4KV BUS 13 MAIN/RES BRKR AUTO CLOSE
DIESEL GEN 1/2 TROUBLE
480V SWGR BREAKER TRIP
BATTERIES UNDERVOLTAGE
4KV BUS 11 MAIN/RES BRKR AUTO CLOSE
4KV BUS 12 MAIN/RES BRKR AUTO CLOSE
4KV BUS 13-1 DIESEL GEN 1/2 BRKR AUTO CL
DIESEL GEN 1/2 AUTO START OR AUTO ST BLOCK
BUS 14 MAIN & RESERVE BRKR PARALLEL
4KV BUS 13-1 VOLTAGE UNBALANCED
UNIT AUX TRANS 11 TROUBLE
RESERVE AUX TRANS 12 TROUBLE
BUS 13 MAIN & RESERVE BKR PARALLEL
DIESEL GEN 1/2 FAIL TO START
4KV BUS 14 MN/RES BRKR AUTO CLOSE
480V BUS 18-19 TIE ACB TRIP
BUS 11 MAIN & RESERVE ACB PARALLEL
BUS 12 MAIN & RESERVE ACB PARALLEL
4KV BUSES 13/14 LOW VOLTAGE
DIESEL GEN 1/2 NEUT VOLT
4KV BUS 14-1 DIESEL GEN BRKR AUTO CLOSE
480V BUS 15-19 DC CONT PWR FAILURE
4KV BUS 11 & 12 LOW VOLTAGE
RESERVE TRANS 12 TRIP
4KV BUSES 13-1/14-1 LOW VOLTAGE
DIESEL GEN 1/2 OVERLOAD
4KV BUS 13-1 VOLTAGE DEGRADED
4KV BUS 14-1 VOLTAGE UNBALANCED
4KV BUS DC POWER FAILURE
RES AUX TRANS 12 SUDDEN PRESS RELAY
4KV BUS OVRCUR TRIP
DIESEL GEN 1/2 LOW FREQUENCY
MAIN XFMR 1 SUDDEN PRESS INHIBIT
GENERATOR 1 GROUND RELAY TROUBLE
4KV TIE BKR BUS 13-1 TRIP
RES AUX TRANS 12 LOW VOLTAGE
4KV TIE BKR BUS 14-1 TRIP
DIESEL GEN 1/2 DAY TANK LEVEL HI/LO
DIESEL GEN 1/2 RELAY TRIP
GEN 1 VOLT REG COMMON ALARM
DIESEL OIL 1/2 STORAGE TANK LOW LEVEL
BUS 11/12 BRKR CLG FAN FAILURE
4KV BUS TIE 13/13-1 14/14-1 DIFF FAULT
OPEN PHASE DETECTION LOW LOAD/TRBLE
DIESEL ROOM 1/2 HIGH TEMP
GEN 1 VOLT REG CONTOL IS LOCAL
POWER DISTRIBUTION
1/2 DIESEL GEN
POWER

Answer: C

Explanation:

A) Incorrect. QCOA 6600-02 required to remedy start failure. Plausible if applicant has a misconception that since the Unit 1 EDG is running that the 1//2 EDG is not required.
B) Incorrect. QCAN 901-8 A7 indicates that there is a problem associated with the Unit 1 EDG that may lead to a failure of that EDG, resulting in a SBO. Plausible if the examinee does not associate the trouble alarm as a possible precursor to the loss of the EDG; some EDG trips are bypassed in emergency situations.
C) Correct. QCAN 901-8 A7 indicates that there is a problem associated with the Unit 1 EDG that may lead to a failure of that EDG, resulting in a SBO A LOOP did occur and the LOOP QCOA contains actions to respond to the to be implemented. The 1/2 EDG did fail to start so its associated QCOA needs to be implemented.
D) Inorrect. The Unit 1 EDG has started and is supplying its associated Busses; therefore a SBO has not occurred. Plausible if the applicant associates the QCAN with the loss or imminent loss of the Unit 1 EDG.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	264000G2.2.44	
Importance Rating		4.4

K/A Statement: Emergency Generators (Diesel/Jet): Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions (CFR: 41.5 / 43.5 / 45.12)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1) QCOA 6100-03 LOSS OF OFFSITE POWER
2) QCOA 6100-04 STATION BLACKOUT
3) QCOA 6600-02 DIESEL GENERATOR 1/2 FAILS TO START
4) QCAN 901-8 A7 DIESEL GEN #1 TROUBLE

Proposed references to be provided to applicants during examination: None

Learning Objective: None

Question Source:

Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	____
	Comprehension or Analysis	X
Level of Difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	____
	55.43	B.1

Comments:

SRO Level justification: Conditions and Limitations in the Facility License. Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations.

Question: 91

Unit 2 is operating at 100% power when annunciator 902-6, F-6 RFP AUTO TRIP alarms.

Thirty seconds later, the following annunciators alarm:

- 902-5, F-8 RX VESSEL LOW LEVEL
- 902-4, F-7 RECIRC LOOPS LIMITED BY FW FLOW/RX LEVEL

Which one of the following best describes the:

(1) impacts on the plant based on the alarms received; and

(2) procedures that should be entered and what direction should the Unit Supervisor provide operators under these conditions?

A. (1) Reactor Recirculation pumps will runback to 32% speed after a 15 second delay

(2) QCOA 3200-01, "Reactor Feed Pump Auto Trip" and perform an emergency power reduction in conjunction with the recirculation runback to maintain reactor level above the low level SCRAM setpoint.

B. (1) Reactor Recirculation pumps will runback to 32% speed after a 15 second delay

(2) QCOA 0400-02, "Core Instabilities" and order a manual speed hold to block the recirculation runback in order to first reduce power with CRAM rod insertion to avoid the instability region.

C. (1) Reactor Recirculation pumps will runback to 65% speed immediately

(2) QCOA 3200-01, "Reactor Feed Pump Auto Trip" and perform an emergency power reduction in conjunction with the recirculation runback to maintain reactor level above the low level SCRAM setpoint.

D. (1) Reactor Recirculation pumps will runback to 65% speed immediately

(2) QCOA 0400-02, "Core Instabilities" and order a manual speed hold to block the recirculation runback in order to first reduce power with CRAM rod insertion to avoid the instability region.

Answer: C

Explanation:

A) Incorrect. (1) Plausible as the anti-cavitation runback causes the recirculation pumps to runback to 32% when feed flow is < 20% for 15 seconds. In this case feed flow will not go below 20% with 2 RFPs running. (2) Operators should initiate an emergency power reduction IAW QCOA 3200-01 and QCGP 3-1 if the runback does not lower power enough to control level and get below the feed pump flow limit for 2 pump operation.

B) Incorrect. (1) Plausible as the anti-cavitation runback causes the recirculation pumps to runback to 32% when feed flow is < 20% for 15 seconds. In this case feed flow will not go below 20% with 2 RFPs running. (2) Plausible since a runback to 32% would place the plant in the instability region without control rod insertion and a manual speed hold would block the runback. Based on plant conditions given, not expected to enter the instability region.

C) Correct. (1) With steam flow > 85% and three RFPs running, if a RFP auto trips and a low reactor water level (26 inches) is attained with 45 seconds of the pump trip, the recirculation pumps will runback to 70% flow (65% speed) in order to prevent a low reactor water level SCRAM. (2) Operators should initiate an emergency power reduction IAW QCOA 3200-01 and QCGP 3-1 if the runback does not lower power enough to control level and get below the feed pump flow limit for 2 pump operation.

D) Incorrect. . (1) With steam flow > 85% and three RFPs running, if a RFP auto trips and a low reactor water level (26 inches) is attained with 45 seconds of the pump trip, the recirculation pumps will runback to 70% flow (65% speed) in order to prevent a low reactor water level SCRAM. (2) Plausible since, with different initial conditions, a runback could place the plant in the instability region without control rod insertion and a manual speed hold would block the runback. Based on plant conditions given, not expected to enter the instability region.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	____	2
Group #	____	2
K/A #	202002 A2.06	
Importance Rating	3.3	

K/A Statement: RECIRCULATION FLOW CONTROL SYSTEM: Ability to (a) predict the impacts of the following on the RECIRCULATION FLOW CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low reactor water level (CFR 43.5)

Technical Reference(s):
(Attach if not previously provided, including version/revision number.)

1. QCOA 3200-01, Reactor Feed Pump Auto Trip, Rev. 23
2. QCOA 0400-02, Core Instabilities, Rev. 26
3. QCAN 901-4, F-7, Recirculation Loop Flows Limited By Feedwater Flow and Reactor Vessel Level, Rev. 06
4. QCGP 3-1, Reactor Power Operations, Rev. 85
5. LN-0202, Reactor Recirculation System
6. LIC-3200, Feed and Condensate

Proposed references to be provided to applicants during examination: None

Learning Objective:	SR-0202-K13		
Question Source:	Bank #	X	
	Modified Bank #	____	
	New	____	
Question History:	Last NRC Exam	DAEC NRC ILT EXAM 2011	
Question Cognitive Level:	Memory or Fundamental Knowledge		____
	Comprehension or Analysis		X
Level of difficulty:	(1-5) 3		
10 CFR Part 55 Content:	55.41 ____		
	55.43 5		

Comments:

Modified to meet QC plant specific nomenclature and procedures.

SRO Only Justification: Requires the assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure to mitigate or recover, or with which to proceed.

Question: 92

Following an inadvertent MSIV closure from 100% power, Unit 2 experienced an ATWS and Loss of Offsite Power.

- QGA 101 RPV CONTROL (ATWS) and QGA 200 PRIMARY CONTAINMENT CONTROL are being implemented.
- Reactor Power is approx. 25% and cycling with relief valve cycles.
- Drywell pressure is 1.5 psig and rising 0.01 psig/minute.
- Suppression Pool temperature is 112˚F and rising 0.5 ˚F/minute.
- ‘B’ RHR Loop is aligned for Torus cooling
- The SRV and all ERVs are cycling
- RPV level is -30 inches and being deliberately lowered per QGA-101

RPV water level has been stabilized at -100 inches.

What actions will be required to ensure that Torus water temperature is reduced?

A. Reclose MO 2-1001-16B, RHR HX BYPASS ONLY.

B. Reopen torus cooling valves TORUS TEST OR SPRAY VLV MO 2-1001- 34B; and TORUS TEST VLV MO 2-1001- 36B ONLY.

C. Align the ‘A’ RHR Loop for Torus Cooling; and
Reclose RHR HX BYPASS VALVES MO 2-1001-16A and 16B.

D. Align the ‘A’ RHR Loop for Torus Cooling; and
Reopen torus cooling valves TORUS TEST OR SPRAY VLV MO 2-1001- 34A and 34B; and TORUS TEST VLV MO 2-1001- 36A and 36B.

Answer: C

Explanation:

A) Incorrect. This action will be necessary to ensure heat removal from the 'B' RHR Loop, since the valve will open when level is dropped below -59", but with Torus temperature above 95°F and rising, the 'A' RHR Loop is needed to maximize cooling.
B) Incorrect. These valves are required to be open for Torus water cooling. However, with the Containment Cooling Permissive switch in "ON", these valves will not shut on the LPCI signal at -59 inches RPV water level. Additionally, the 'A' RHR Loop is also needed to maximize cooling.
C) Correct. With Torus temperature above 95°F and rising, the 'A' RHR Loop is needed to maximize cooling. The HX bypass valves will open when level is dropped below -59" and will need to be reclosed if open.
D) Incorrect. With Torus temperature above 95°F and rising, the 'A' RHR Loop is needed to maximize cooling. The Torus cooling valves are required to be open for Torus water cooling. However, with the Containment Cooling Permissive switch in "ON", these valves will not shut on the LPCI signal at -59 inches RPV water level. .

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	____	2
Group #	____	2
K/A #	219000G2.1.7	
Importance Rating	____	4.7

K/A Statement: RHR/LPCI: Torus/Suppression Pool Cooling Mode: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation (CFR: 41.5 / 43.5 / 45.12 / 45.13))

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. LN-1000 Residual Heat Removal, Rev. 20
2. QGA-101 RPV Control (ATWS), Rev. 15
3. L-QGA101 QGA 101, RPV Control (ATWS), Rev. 11
4. QCOA 1600-03 Torus Water High Temperature, Rev. 14

Proposed references to be provided to applicants during examination: None

Learning Objective: S-0001-K061 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.

Question Source:

Bank #	_______	
Modified Bank #	X	(Note changes or attach parent)
New	_______	

Question History: Last NRC Exam Fitzpatrick 2008 #92

Question Cognitive Level:	Memory or Fundamental Knowledge	____
	Comprehension or Analysis	X
Level of Difficulty:	(1-5) 3	
10 CFR Part 55 Content:	55.41 5	
	55.43 5	

Comments:

Adapted a draft question from the 2008 Fitzpatrick exam to meet NUREG 1021 standards and Quad Cities terminology.

SRO Level Justification: Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations.

Question: 93

Unit 2 reactor SCRAMS from 100% power.

- All rods are fully inserted.
- Reactor pressure is 920 psig
- 902-3, C-2, OFF GAS HIGH HIGH RADIATION is in alarm
- DW GAMMA RADIATION, RR 2-2420-A is reading 2000 R/hr

Which of the following is the required Emergency Classification?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: D

Explanation:

A) Incorrect. Plausible as the OFF GAS Hi Hi alarm is a separate EAL entry criteria for RU3, if the high DW rads were not considered.
B) Incorrect. Plausible since without evidence of a leak, the applicant may determine that only the fuel clad fission product barrier has failed.
C) Incorrect. Plausible since the applicant may not recognize that the rad level indicates a loss or potential loss of all three fission product barriers.
D) Correct. DW radiation monitor reading 2000 R/hr would indicate a loss or potential loss of all three fission product barriers, resulting in a FG1 EAL declaration.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	____	2
Group #	____	2
K/A #	272000 2.4.21	
Importance Rating	4.6	

K/A Statement: RADIATION MONITORING SYSTEM: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (CFR 43.5)

Technical Reference(s): EP-AA-1006, Addendum 3, Emergency Action Levels for Quad Cities Station, Rev. 4

Proposed references to be provided to applicants during examination:

EP-AA-1006, Addendum 3, Emergency Action Levels for Quad Cities Station, Rev. 4

Learning Objective: S-1701-K70

Question Source:
Bank # __
Modified Bank # __
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge __
Comprehension or Analysis X

Level of difficulty: (1-5) 2

10 CFR Part 55 Content:
55.41 ___
55.43 2 and 5

Comments:

SRO Only Justification: Requires analysis and interpretation of radiation readings as they pertain to emergency plan criteria.

Question: 94

Complete the following two statements regarding interpretation and execution of Quad Cities Emergency Operating Procedures (QGAs):

(1) Per the QGA Marking Standards, an ARROW pointing at a step in a QGA leg indicates that the crew …..

(2) Per OP-QC-103-102-1002, Quad Cities Strategies for Successful Transient Mitigation, when executing a leg of the QGAs, all steps should be …..

A. (1) has completed the referenced step
   (2) followed in order, even if an emergency depressurization parameter is exceeded further down in the leg

B. (1) has completed the referenced step
   (2) omitted up to the blowdown step if an emergency depressurization parameter is exceeded further down in the leg

C. (1) is maintaining or waiting for a specific plant condition
   (2) followed in order, even if an emergency depressurization parameter is exceeded further down in the leg

D. (1) is maintaining or waiting for a specific plant condition
   (2) omitted up to the blowdown step if an emergency depressurization parameter is exceeded further down in the leg

Answer: C

Explanation:

When actions have progressed to a point where the crew is maintaining or waiting for a specific plant condition, an arrow should be used to mark their place.

When executing a leg of the QGA's, all steps should be followed in order to allow the use of all of the mitigating systems to be used and their effectiveness assessed, even if a blowdown parameter is exceeded further down in the leg.

A) Incorrect, plausible because there is a marking standard to indicate a step is complete (line through step)
B) Incorrect, combination of A and D
C) Correct
D) Incorrect, plausible because a blowdown is typically executed to prevent the plant from violating an unsafe condition.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		3
Group #		-
K/A #	G2.1.20	
Importance Rating		4.6

K/A Statement: Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QCAP 0200-10, EMERGENCY OPERATING PROCEDURE (QGA) EXECUTION STANDARDS; Rev 54
2. OP-QC-103-102-1002, Quad Cities Strategies for Successful Transient Mitigation; Rev 22

Proposed references to be provided to applicants during examination: None

Learning Objective: S-000-K03-DESCRIBE the QGA flowchart procedure structure.

Question Source:

Bank #		
Modified Bank #		(Note changes or attach parent)
New	X	

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	___
Level of Difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	___
	55.43	B.5

Comments:

SRO Level Justification: Execution standards of the EOPs are a function of the SRO (see facility objective)>

Question: 95

Unit 2 is in Mode 1.

QCOS 0203-07, “Unit 2 Online Automatic Blowdown Logic Test” is in progress.

- Channel 287-120B of the ‘B’ Electromatic Relief Valve reactuation time delay circuit tested outside of its acceptance band of 10.5 sec ≤ t ≤ 17.8 sec at 7 seconds.
- Channel 287-121B tested satisfactorily at 14.5 seconds.

(1) What is the purpose of the reactuation time delay circuit?

AND

(2) What actions, if any, are required per the Technical Specifications for channel 287-120B testing out of the allowable band?

A. (1) Establish a relief valve opening sequence which allows for a balanced heating pattern of the Torus suppression pool.
(2) Enter multiple 14 day LCO actions for an inoperable ‘B’ relief valve.

B. (1) Allow the high water leg created from initial valve cycle to return to its normal level; thus, reducing thrust loads from subsequent actuations to within design limits.
(2) NO LCO action is required, since one channel of reactuation time delay is functioning properly.

C. (1) Establish a relief valve opening sequence which allows for a balanced heating pattern of the Torus suppression pool.
(2) NO LCO action is required, since one channel of reactuation time delay is functioning properly.

D. (1) Allow the high water leg created from initial valve cycle to return to its normal level; thus, reducing thrust loads from subsequent actuations to within design limits.
(2) Enter multiple 14 day LCO actions for an inoperable ‘B’ relief valve.

Answer: D

Explanation:

A) Incorrect. Plausible since balanced heating of the Torus is desirable. 2nd part is correct.
B) Incorrect. Plausible as part 1 is correct. Part 2 is incorrect because TS 3.3.6.3 requires both channels to be operable.
C) Incorrect. Plausible since balanced heating of the Torus is desirable. Part 2 is incorrect because TS 3.3.6.3 requires both channels to be operable.
D) Correct. In order to ensure the containment structure is not subject to excessive stress from a large water slug in the relief valve piping following the initial actuation, the low set valves (B&C) have a reactuation time delay of 10 to 17 seconds. This allows enough time for the vacuum breaker on the discharge pipe to enable the higher column of water to drain back to the suppression pool before the relief valve reopens. Both channels of reactuation time delay are required to be operable as the logic is 2 out of 2. Therefore must enter 14 day LCO actions for TS 3.3.6.3, Relief Valve Instrumentation; TS 3.6.1.6, Low Set Relief Valves; TS 3.4.3, Safety and Relief Valves; and TS 3.5.1, ECCS Operating.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	2.1.28	
	Importance Rating	4.1	

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

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QCOS 0203-07, Unit 2 Online Automatic Blowdown Logic Test, Rev. 19
2. Technical Specification Bases B3.3.6.3, Relief Valve Instrumentation

Proposed references to be provided to applicants during examination: None

Learning Objective: S-0203-K33 and S-0250-K33

Question Source:
- Bank # ______
- Modified Bank # ______ (Note changes or attach parent)
- New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	___
	Comprehension or Analysis	_X_
Level of difficulty:	(1-5)	__3__
10 CFR Part 55 Content:	55.41	___
	55.43	_2_

Comments:
SRO Only Justification: Knowledge of required actions as derived from the Technical Specifications. Knowledge of TS Bases that is required to analyze TS-required actions and terminology.

Question: 96

Unit 1 is at 100% power.

- HPCI has been removed from service for corrective maintenance and declared INOPERABLE.
- An Equipment Outage Report was generated in accordance with QCAP 0230-19, “EQUIPMENT OPERABILITY”.

Which of the following conditions are required to return HPCI to OPERABLE status at the conclusion of maintenance?

1) All issue reports documenting discrepancies identified during maintenance must be closed.

2) Operability Determinations must be completed and approved for all issue reports documenting discrepancies identified during maintenance.

3) Components that reposition or change state during an automatic initiation need not be in a Standby lineup provided all required power supplies are available to support auto initiation.

A. 1 only

B. 2 only

C. 2 and 3

D. 1 and 3

Answer: B

Explanation:

A) Incorrect. The issue report need not be closed, only an Operability Determination must be completed. Plausible since issue reports are used to identify discrepancies that may impact operability.

B) Correct. If the LCO condition involves an Issue Report documenting a degraded or nonconforming condition, then an approved Operability Determination exist prior to clearance of the LCO.

C) Incorrect. Statement 3 is incorrect. The system is required to be restored to a standby lineup or as determined by the Shift Manager to be acceptable to the current mode. Since the plant is at rated power conditions, HPCI would be placed in a standby lineup. Plausible since required components are interlocked to realign to their required position upon auto initiation.

D) Incorrect. The Issue report need not be closed. Only an Operability Determination must be completed. Also HPCI is required to be in a standby lineup. See A and C for plausibility.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	____	3
Group #	____	____
K/A #	G2.2.21	
Importance Rating	____	4.1

K/A Statement: Knowledge of pre- and post-maintenance operability requirements. (CFR: 41.10 / 43.2)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. L-OPDT Operability Determinations, Rev. 8
2. OP-AA-108-115 Operability Determinations (CM-1), Rev. 19
3. OP-AA-108-115-1002 Supplemental Considerations for On-Shift Immediate Operability Determinations, Rev. 3
4. QCAP 0230-19, “Equipment Operability”; Rev 19

Proposed references to be provided to applicants during examination:

OP-AA-108-115 and OP-AA-108-115-1002

Learning Objective:

Given a copy of the Operability Determination procedures, OP-AA-108-115 and OP-AA-108-115-1002, DESCRIBE the following items related to Operability Determination.

a. When a procedure is used as justification in an operability evaluation or for compensatory action.

b. When work orders are generated as part of an operability evaluation corrective action.

c. When operability evaluations should be revised.

d. The use of Engineering judgment.

e. The use of PRA.

f. The use of testing.

Question Source:	Bank #	X	
	Modified Bank #	________	(Note changes or attach parent)
	New	________	
Question History:	Last NRC Exam	Pilgrim 2013 #95	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis	_____	
Level of Difficulty:	(1-5)	3	
10 CFR Part 55 Content:	55.41	_____	
	55.43	B.2	

Comments:

SRO Level Justification: Facility Operating Limitations in the Technical Specifications and their Basis.

Question: 97

0900 A liquid release to the river begins IAW QOP 2000-25, Attachment A, “Liquid Radioactive Waste Discharge Sheet Discharging the River Discharge Tank With the River Discharge Pump.”

Plant conditions at the start of the release:

- Ice melt isolation valve open
- All circulating water (CW) pumps running for both units and the 1A, 2A, and 1B service water pumps running (minimum running dilution pumps determined to be 5 CW pumps and 3 service water pumps)
- River Discharge Tank (RDT) Level 60%

1100 This discharge was temporarily halted.

- RDT Level is 50%

1200 The discharge is set to recommence under the same Discharge Sheet. Several items have changed since the discharge was temporarily stopped:

- Ice melt isolation valve was CLOSED
- 2A CW Pump was stopped
- RDT Level is 55%
- ½-2002-93, Radwaste Liquid Effluent Flow Rate Recorder has failed low

Which of the following conditions requires the Liquid Radioactive Waste Discharge Sheet to be voided and re-performed prior to commencing the RDT release to the river?

A. 2A CW stopped

B. 1/2 -2002-93, Radwaste Liquid Effluent Flow Rate Recorder failed low

C. Ice melt isolation valve CLOSED

D. RDT Level increasing from 50% to 55%

Answer: D

Explanation:

A) Incorrect. Plausible as stopping a CW pump reduces dilution flow, but the original Discharge Sheet indicated minimum pumps for dilution flow was 5 CW and 3 service water. When the 2A CW pump was stopped, the minimum number of dilution pumps were running IAW the original Discharge Sheet.
B) Incorrect. Plausible as the Radwaste Liquid Effluent Flow Rate Recorder is relied upon to provide pertinent information about the volumetric flow rate of the release. QOP 2000-25 allows the release to continue without this instrument if the licensee takes action to periodically calculate flow rate based on other parameters and works to restore the instrument.
C) Incorrect. Plausible as changing the position of the ice melt valve in the open direction would require dilution flow to be recalculated on the discharge sheet before release can be restarted. Changing the ice melt valve in the closed direction is conservative with regards to dilution flow.
D) Correct. If level of the RDT increases by more than 2% during a release or during a temporary hiatus in a release the Discharge Sheet must be voided and the tank resampled to ascertain the composition of the new liquid which was added to the tank to ensure the release will be within federal regulations.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	2.3.6	
	Importance Rating	3.8	

K/A Statement: Ability to approve release permits (CFR 43.4)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QOP 2000-25, Discharging to the River from the River Discharge Tank Using the River Discharge Pump, Rev. 47
2. LN-2000, Radioactive Waste Processing

Proposed references to be provided to applicants during examination:

QOP 2000-25, Discharging to the River from the River Discharge Tank Using the River Discharge Pump

Learning Objective: S-2001-K35

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	___
	Comprehension or Analysis	X
Level of difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	____
	55.43	4

Comments:

SRO Only Justification: Radiation hazards that may arise during normal and abnormal situations including maintenance activities and various contamination conditions: Process for gaseous/liquid release approvals.

Question: 98

(1) The Drywell Radiation Monitors [RE-1(2)-2418A/B] indicate _________.

(2) The Drywell Gamma Radiation recorders [RR-1(2)-2420A/B] _________ required to be operational to support OPERABILITY of the Post Accident Monitoring (PAM) Instrumentation Drywell Radiation Monitors function.

A. (1) continuously during all modes of plant operation
(2) are

B. (1) only during post-accident conditions
(2) are

C. (1) continuously during all modes of plant operation
(2) are NOT

D. (1) only during post-accident conditions
(2) are NOT

Answer: A

Explanation:

A) Correct: (1) The monitors provide continuous indication of the radiation levels in the Drywell. The recorders are actuated upon detection of a LOCA (Core Spray Initiation). (2) Per the bases for Technical Specification LCO 3.3.3.1, the recorder function must operational for the Drywell Radiation Monitors function to be considered OPERABLE.
B) Incorrect: (1) The monitors provide continuous indication, however the recorders are NOT activated until a LOCA condition is detected (Core Spray Initiation). (2) Per the bases for Technical Specification LCO 3.3.3.1, the recorder function must operational for the Drywell Radiation Monitors function to be considered OPERABLE.
C) Incorrect: (1) The monitors provide continuous indication of the radiation levels in the Drywell. The recorders are actuated upon detection of a LOCA (Core Spray Initiation). (2) Since the recorders are not continuously in operation, the examinee may not believe that the recorders are required to satisfy operability.
D) Incorrect: (1) The monitors provide continuous indication, however the recorders are NOT activated until a LOCA condition is detected (Core Spray Initiation). (2) Since the recorders are not continuously in operation, the examinee may not believe that the recorders are required to satisfy operability.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		3
Group #		
K/A #	G2.3.15	
Importance Rating		3.1

K/A Statement: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.
(CFR: 41.12 / 43.4 / 45.9)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

(1) LN 2400, Containment Atmosphere Monitoring (CAM)
(2) Technical Specification LCO 3.3.3.1, Post Accident Monitoring (PAM) Instrumentation and associated Bases

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____
Level of Difficulty:	(1-5) __3__	
10 CFR Part 55 Content:	55.41(a) _____	
	55.43(b) _2&4_	

Comments: SRO justification: knowledge of TS bases that is required to analyze TS-required actions and terminology, Knowledge of radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Question: 99

A large break LOCA occurred on Unit 1.

- RPV pressure is 15 psig and slowly lowering.
- 3 control rods remain at position 04, all others are full in.
- DW pressure is 15 psig and slowly rising.
- DW temperature is 250ºF and slowly rising.
- Torus temperature is 180 ºF and slowly rising.
- Narrow range RPV water level indications were downscale but are now trending up.
- Medium range RPV water level indications were downscale but are now trending up.
- Fuel Zone and Lower Wide Range RPV water level indications had been indicating at approximately -160 inches but are now behaving erratically with a general upward trend.
- There has been no measurable increase in RPV injection rates.

Which of the following actions is required to be directed?

A. Establish a known RPV level by performing QGA 500-4, "RPV Flooding"

B. Shutdown the reactor by performing QGA 101, "RPV Control (ATWS)"

C. Ensure adequate core cooling by performing QGA 500-2, "Steam Cooling"

D. Cool the Drywell by initiating Drywell Sprays as directed by QGA 200 "Primary Containment Control."

Answer: A

Explanation:

A) Correct. With containment conditions adverse and no viable level indication, all EOPs and contingency procedures direct operators to establish a known RPV level with QGA 500-4, “RPV Flooding”.
B) Incorrect. Plausible as there are multiple control rods not full inserted. QGA 100, “RPV Control” indicates that the reactor is shutdown under all conditions with all rods inserted to at least the 04 position.
C) Incorrect. Plausible as QGA 100, “RPV Control” directs operators to QGA 500-2, “Steam Cooling” before level reaches the MSCWL with no injection sources running. In this case adverse conditions in containment have resulted in unreliable RPV level indication and there is no mention in the question stem that RPV injection sources do not exist.
D) Incorrect. Plausible as QGA 200, “Primary Containment Control” would direct spraying of the Drywell due to Drywell pressure and DW temperature, but all EOPs and Contingency EOPs direct entry to QGA 500-4, “RPV Flooding” when RPV water level is unknown. With RPV level unknown, adequate core cooling cannot be assured.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	____	3
Group #	____	1
K/A #	2.4.6	
Importance Rating	4.7	

K/A Statement: Knowledge of EOP mitigation strategies (CFR 43.5)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

1. QGA 100, RPV Control, Rev. 11
2. QGA 101, RPV Control (ATWS), Rev. 15
3. QGA 500-1, RPV Blowdown, Rev. 15
4. QGA 500-2, Steam Cooling, Rev. 11
5. QGA 500-4, RPV Flooding, Rev. 14
6. QGA 200, Primary Containment Control; Rev 11

Proposed references to be provided to applicants during examination:

Detail A: RPV Water Level Instruments from QGA 100, RPV CONTROL

Learning Objective: S-0001-K56

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	_
	Comprehension or Analysis	X
Level of difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	____
	55.43	5

Comments:
SRO Only Justification: Assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed.

Question: 100

During the implementation of EOPs, safety-related systems/components...

A. can be disabled ONLY AFTER notifying the NRC of entry into 10 CFR 50.54(x).

B. are NEVER intentionally disabled.

C. are disabled, in accordance with the EOPs/EOP Implementing procedures, when directed by the Unit Supervisor.

D. can be disabled ONLY AFTER receiving approval from the Site Emergency Director in the TSC.

Answer: C

Explanation:

The EPGs/ authorize defeating specific trips, isolations, interlocks, and initiation logic that may interfere with EPG/SAG objectives, provided the associated protective functions are unnecessary or inappropriate under the conditions considered or are addressed through prescribed operator actions. The specified actions are performed in accordance with approved procedures, directed by the Unit Supervisor, and do not require declaration of 10CFR 50.54(x) or additional approval or authorization.

A) Incorrect: Plausible since 10 CFR 50.54(x) and (y) allow reasonable actions that depart from licensed conditions or technical specifications, during emergency conditions, but do not require prior NRC notification.
B) Incorrect: Plausible safety-related systems/components are not disabled during normal operation unless permitted within technical specification guidance.
C) Correct: See above
D) Incorrect: Plausible since certain post-accident actions are permitted only with authorization by the position with Command Control during an Emergency.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	___	3
Group #	___	___
K/A #	G2.4.14	
Importance Rating	___	4.5

K/A Statement: Knowledge of general guidelines for EOP usage. (CFR: 41.10 / 43.1 / 45.13)

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
1) QCAP 0200-10, Emergency Operating Procedure (QGA) Execution Standards, Rev. 54
2) BWR Owners' Group Emergency Procedure and Severe Accident Guidelines; Rev 3
3) 10 CFR 50.54(x) and (y)

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # X
Modified Bank # _______ (Note changes or attach parent)
New _______

Question History: Last NRC Exam Quad Cities 2005 #99(SRO 24)

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
Level of Difficulty:	(1-5)	3
10 CFR Part 55 Content:	55.41	
	55.43	B.1

Comments:

SRO Justification: Conditions and Limitations in the Facility License