Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295001 AK2.02	
	Importance Rating	3.2	

K/A Statement: Knowledge of the interrelations between partial or complete loss of forced core flow circulation and the following: Nuclear boiler instrumentation.

Proposed Question: **Question #1**

Unit 1 was operating at 100% RTP when the 1A Reactor Recirculation Pump tripped.

When conditions stabilize, with NO operator actions:

1) Flow in the 'A' Loop is _______.

2) Indicated Total Core Flow will be _______.

A. (1) forward
(2) equal to actual core flow

B. (1) reverse
(2) equal to actual core flow

C. (1) forward
(2) greater than actual core flow

D. (1) reverse
(2) greater than actual core flow

Proposed Answer: B

Explanation:

A) Incorrect. Plausible because flow will continue to exist in the idled loop, but differential pressure conditions from the operating loop will cause flow to be reversed. Part 2 is correct.
B) Correct. Flow through the operating loop will drive reverse flow conditions in the idle loop. Total core flow will indication will be the actual core flow due to the Forward-Reverse logic circuit that does not add reverse flow from the jet pumps to the total core flow.
C) Incorrect. Plausible because flow will continue to exist in the idled loop, but differential pressure conditions from the operating loop will cause flow to be reversed. Part 2 is plausible as without the Forward-Reverse logic circuit, indicated total core flow would include reverse flow through the idled RR loop and therefore would be greater than actual total core flow.
D) Incorrect. Plausible as Part 1 is correct, and without the Forward-Reverse logic circuit, indicated total core flow would include reverse flow through the idled RR loop and therefore would be greater than actual total core flow.

Technical Reference(s): 022 Reactor Recirculation Rev 17 (p. 58), 040 Rx Vessel Inst. Rev 8 (p. 12), LOA-RR-101 Rev 42

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 040.00.07

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

Question History: Last NRC Exam NA

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis 2

10 CFR Part 55 Content:
55.41 7
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295003 AK3.06	
	Importance Rating	3.7	

Knowledge of the reasons for the following responses or actions as they apply to partial or complete loss of AC power: Containment Isolation

Proposed Question: **Question #2**

Unit 1 is at 100% RTP when a loss of RPS Bus B power occurs.

How will the following valves respond?

	1G33-F001 RWCU Suction Inboard Valve	1G33-F004 RWCU Suction Outboard Valve
A.	OPEN	CLOSED
B.	CLOSED	OPEN
C.	OPEN	OPEN
D.	CLOSED	CLOSED

Proposed Answer: D

Explanation:

A) Incorrect. Plausible as a loss of RPS A will cause a closure of the outboard isolation valve only for certain groups of containment isolation.
B) Incorrect. Plausible because other Groups of containment isolation logic will close the inboard isolation valve only on a loss of RPS B.
C) Incorrect. Plausible if the applicant believes the loss of RPS B power will also affect power to operate the Group 5 containment isolation MOVs.
D) Correct. De-energizing the RPS B Division 2 inboard isolation logic will cause all inboard and outboard isolation valves for Group 5 RWCU (and other Groups) to close. This failure affects logic power only and not the MOV power.

Technical Reference(s): 091 PCIS Rev 15 (p. 41)

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 091.00.16

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

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge 3
Comprehension or Analysis __

10 CFR Part 55 Content: 55.41 5
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295004 AA1.02	
	Importance Rating	3.8	

K/A Statement: Ability to operate and/or monitor the following as they apply to partial or complete loss of DC power: Systems necessary to assure safe plant shutdown

Proposed Question: **Question #3**

At time 0700, Unit 2 was operating at 80% RTP when the following occurred:

- 250 VDC voltage reads 0 V at 2PM01J, and the undervoltage alarm is in
- 125 VDC Div 2 voltage reads 0 V at 2PM01J, and the undervoltage alarm is in
- Reactor was manually scrammed IAW LOA-DC-201, “Unit 2 DC Power System Failure”

At time 0705, Unit 2 experiences a loss of cooling accident and loss of offsite power.

- RPV pressure is 440 psig and lowering

Which of the following will be available for the crew to maintain RPV level from the control room?

A. LPCS Pump

B. ‘B’ CRD Pump

C. ‘C’ LPCI Pump

D. RCIC Pump

Proposed Answer: A

Explanation:

A) Correct. With a loss of offsite power on Unit 2, the 0 EDG will automatically start and re-energize the 241Y 4-kV bus. The LPCS pump is powered from the 241Y 4-kV bus and therefore will be available as RPV pressure is below the shutoff head (470 psig) of the pump.
B) Incorrect. Plausible as the 'B' CRD Pump would normally be available when the 2A EDG automatically starts and re-energizes the 242Y 4-kV bus on a loss of offsite power on Unit 2. Due to the fault on the 125 VDC Div 2 Bus, the 2A EDG will not automatically start due to loss of field flash. With no additional operator action, the 'B' CRD Pump will not have power.
C) Incorrect. Plausible as the 'C' LPCI Pump would normally be available when the 2A EDG automatically starts and re-energizes the 242Y 4-kV bus on a loss of offsite power on Unit 2. Due to the fault on the 125 VDC Div 2 Bus, the 2A EDG will not automatically start due to loss of field flash. With no additional operator action, the 'C' LPCI Pump will not have power. This distractor is also plausible if the examinee mistakes the 'C' LPCI Pump for Div 1.
D) Incorrect. Plausible as the RCIC pump uses steam as the motive force and the examinee may consider the RCIC pump will still run with a loss of offsite power. However, with the failure of the 250 VDC Bus 2, the RCIC injection valve, 2E51-F013 will fail to open, so RCIC will be unable to maintain RPV level.

Technical Reference(s): 006 DC Rev 10 (p. 2), LOA-DC-201 Rev 21, LOA-AP-201 Rev 53, LOP-AP-242Y Rev 15

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 006.00.18

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

Question History: Last NRC Exam NA

Question Cognitive Level:	Memory or Fundamental Knowledge	____
	Comprehension or Analysis	3

10 CFR Part 55 Content: 55.41 7
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295005 AA2.06	
	Importance Rating	2.6	

K/A Statement: Ability to determine and/or interpret the following as they apply to main turbine generator trip: Feedwater Temperature

Proposed Question: **Question #4**

The plant is operating at 22% RTP, with the Feedwater heaters on-line, when an unknown failure causes a main turbine trip.

How will reactor power respond to this transient?

A. Reactor power remains constant because feedwater temperature remains constant.

B. Reactor power lowers because the reactor scrams from the main turbine trip.

C. Reactor power rises due to lowering feedwater temperature.

D. Reactor power lowers due to rising feedwater temperature.

Proposed Answer: C

Explanation:

A) Incorrect. Plausible if the applicant does not recall that a turbine trip results in a loss of extraction steam to the feedwater heater and hence a lowering in feedwater temperature.
B) Incorrect. Plausible if the applicant determines that the reactor will scram on a turbine trip at 22% power. The reactor scrams if power is greater than or equal to 25% at the time of a turbine trip.
C) Correct. At 22% power, the reactor does not scram from a turbine trip. The trip causes a loss of extraction steam to the feedwater heaters and a corresponding decrease in feedwater temperature to the reactor. Reactor power increases from the decrease in feedwater temperature.
D) Incorrect. Plausible if the applicant determines that a turbine trip results in a greater amount of extraction steam being sent to the feedwater heaters.

Technical Reference(s): LOA-TG-101 Rev 101, LOR-1H13-P603-A211 Rev 9

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 071.00.10

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

Question History: Last NRC Exam 2008 LaSalle ILT Exam

Question Cognitive Level: Memory or Fundamental Knowledge __
Comprehension or Analysis 2

10 CFR Part 55 Content: 41.7 2-9
45.8 ____

Comments: Enhancements were made to the question but were not enough to consider the question modified.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295006 G2.4.04	
	Importance Rating	4.5	

K/A Statement: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for Emergency and Abnormal Operating Procedures: Scram.

Proposed Question: **Question #5**

Unit 1 was at 100% RTP when the following occurred:

- Drywell pressure is 2.0 psig and RISING
- The crew scrammed the reactor and placed the mode switch to SHUTDOWN
- One rod is at position 48, one rod is at position 04, the remainder of the rods are at position 00

Which of the following must be performed FIRST?

A. IAW LGA-010, “Failure to Scram” inhibit ADS

B. IAW LGA-001, “RPV Control” prevent LPCS and LPCI injection

C. IAW LGA-MS-101, “Unit 1 Using Main Condenser as a Heat Sink in ATWS” bypass MSIV isolations

D. IAW LGA-003, “Primary Containment Control” start suppression chamber sprays

Proposed Answer: A

Explanation:

A) Correct. The crew meets entry conditions for LGA-001 because drywell pressure is above 1.93 psig. Once the scram buttons are depressed and the mode switch is taken to SHUTDOWN, more than one rod did not fully insert. Therefore, next step in LGA-001 is to transition to LGA-010 and inhibit ADS.
B) Incorrect. Plausible, LGA-001 is initially entered and drywell pressure is above 1.77 psig, which meets the criteria to control RPV pressure and level. If the examinee believes the scram is complete/successful and the reactor is shutdown, then controlling RPV pressure and level is the next step to be performed.
C) Incorrect. Plausible because the first step in the LGA-010 Level Leg is to perform LGA-MS-101.
D) Incorrect. Plausible because entry conditions are met for LGA-003 since drywell pressure is above 1.93 psig. However, with the reactor not shutdown, reactivity control is the highest priority, and the actions of LGA-010 to inhibit ADS must be performed first.

Technical Reference(s): LGA-001 Rev 18, LGA-003 Rev 18, LGA-010 Rev 18, LGP-3-2 Rev 74, LGA-MS-101 Rev 3

Proposed references to be provided to applicants during examination: N/A

Learning Objective: ___

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

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis 3

10 CFR Part 55 Content: 55.41 6
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295016 AK2.01	
	Importance Rating	4.4	

K/A Statement: Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: Remote shutdown panel

Proposed Question: **Question #6**

Following the abandonment of the control room, the placing of a component's Transfer Switch to the EMERGENCY position on the Remote Shutdown Panel (RSP) will allow control of the component from the RSP ___(1)___ to enable bringing the reactor to cold shutdown and the component will ___(2)___ automatic primary containment isolation features.

A. (1) AND the control room
(2) maintain

B. (1) AND the control room
(2) lose

C. (1) ONLY
(2) maintain

D. (1) ONLY
(2) lose

Proposed Answer: D

Explanation:

A) Incorrect. In the EMERGENCY position all power is removed from the control room components. Logic for automatic features for most but not all valves is lost because moving the Transfer Switch replaces the normal circuitry with alternate logic circuitry. Trips and interlocks are also bypassed in the EMERGENCY position except for RCIC turbine mechanical overspeed trip, RCIC turbine local mechanical trip, and the RHR suction valve interlock. These exceptions act as good distractors. Neither of the conditions are true.
B) Incorrect. Moving the Transfer Switch replaces the normal circuitry with alternate logic circuitry and all control in the main control room will be lost. The first aspect is false but the second aspect is true.
C) Incorrect. Logic for automatic features for most valves is lost because the Transfer Switch brings in alternate logic circuitry. The first aspect is true but the second aspect is false.
D) Correct. Placing a component's RSP Transfer Switch in EMERGENCY will cause a loss of the associated components control from the control room and will override automatic and protective features of the associated components. Both aspects of this answer are true.

Technical Reference(s): LOP-RX-03 Rev 7, 054 Remote Shutdown System Rev 9 (p. 20)

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 054.00.05c

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 2
Comprehension or Analysis _

10 CFR Part 55 Content: 55.41 (b)(7)

Comments: The bank question was modified to change the location of the correct answer; to add detail to the explanations; and to add an objective. This does not meet the definition of a modified bank question.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295018 AK3.07	
	Importance Rating	3.1	

K/A Statement: Knowledge of the reasons for the following responses as they apply to partial or complete loss of component cooling water: Cross-connecting with backup systems.

Proposed Question: **Question #7**

Unit 1 was at 100% RTP when the following occurred:

- 1PM10J-A304, TBCCW DISCH HDR PRESS LO is LIT
- The crew enters and performs the immediate actions in LOA-WT-101, “Loss of Turbine Building Closed Cooling Water”

Subsequently,

- Both TBCCW Pumps are verified running
- 1PM10J-A304, TBCCW DISCH HDR PRESS LO is still LIT

Based on the above conditions, the crew will ____(1)____ in order to ____(2)____.

A. (1) start a standby Service Water Pump
(2) ensure availability of the Station Air Compressors

B. (1) start a standby Service Water Pump
(2) provide adequate cooling to the MDRFP

C. (1) cross-tie TBCCW from Unit 2
(2) ensure availability of the Station Air Compressors

D. (1) cross-tie TBCCW from Unit 2
(2) provide adequate cooling to the MDRFP

Proposed Answer: C

Explanation:

A) Incorrect. Plausible because the crew will start a standby Service Water Pump for degraded TBCCW flow IAW LOA-WT-101 Section B.1. However, the conditions indicate there is a complete loss of TBCCW flow, which will direct the crew to cross-tie TBCCW from Unit 2 IAW LOA-WT-101 Section B.2. The second part is correct, according to Training Document “113 Turbine Building CCW”, the primary concern with a loss of TBCCW is to ensure the availability of Station Air.

B) Incorrect. Plausible because the crew will start a standby Service Water Pump for degraded TBCCW flow IAW LOA-WT-101 Section B.1. However, the conditions indicate there is a complete loss of TBCCW flow, which will direct the crew to cross-tie TBCCW from Unit 2 IAW LOA-WT-101 Section B.2. The second part is also plausible because the TBCCW system provides cooling for the MDRFP, but with a complete loss of TBCCW flow, the primary concern is the availability of Station Air. There is no mention of cross-tying in order to restore cooling to the MDRFP.

C) Correct. The conditions indicate a total loss of TBCCW, which will direct the crew to LOA-WT-101 Section B.2. The crew will cross-tie TBCCW from Unit 2 per the procedure, and Training Document “113 Turbine Building CCW” states the primary concern with a loss of TBCCW is to secure availability of Station Air.

D) Incorrect. The first part is correct, the conditions indicate a loss of TBCCW, which will direct the crew to LOA-WT-101 Section B.2. The crew will cross-tie TBCCW from Unit 2. The second part is plausible because the TBCCW system provides cooling for the MDRFP, but with a loss of TBCCW flow, the primary concern is the availability of Station Air.

Technical Reference(s): LOA-WT-101 Rev 5, LOR-1PM10J-A304 Rev 2, 113 Turbine Building CCW Rev 6 (p. 2), 120 Plant Air System Rev 12 (p. 24)

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 113.00.18

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

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis 2

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

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295019 AA1.03	
	Importance Rating	3.0	

K/A Statement: Ability to operate and/or monitor the following as they apply to a partial or complete loss of instrument air: instrument air compressor power supplies

Proposed Question: **Question 8**

Unit 1 was operating at 100% RTP:

- Unit 2 Station Air Compressor is OOS
- A scram occurs on Unit 1

Subsequently:

- A Loss of Offsite Power occurs affecting Unit 1 ONLY
- The '1A' EDG does NOT start

Per LOA-AP-101, "Unit 1, AC Power System Abnormal" what actions are required to restore the Common Station Air Compressor to Unit 1?

A. Restore power to the '0' Station Air Compressor, 0SA01C by aligning it at its Unit 2 power supply at Bus 242X

B. Restore power to Bus 142Y by closing the Unit Cross Tie Breakers ONLY

C. Restore power to Bus 142Y by closing the Unit Cross Tie Breakers and then close the 142Y-142X Bus Tie Breaker to restore power to Bus 142X ONLY

D. Restore power to Bus 142Y by closing the Unit Cross Tie Breakers, install jumper to defeat 142Y-142X Bus Tie Breaker interlocks, and then close the 142Y-142X Bus Tie Breaker to restore power to Bus 142X

Proposed Answer: ___D___

Explanation:

A) Incorrect. Plausible as many Common components can be powered from both Unit 1 and Unit 2. The Common Station Air Compressor 0SA01C is only powered from Bus 142X.

B) Incorrect. Plausible as Bus 142Y will need to be energized first and other subset busses are immediately energized when this occurs (e.g. 480VAC ESF busses).

C) Incorrect. Plausible as Bus 142X will need to be restored by cross tying power between the Units and closing the 142Y-142X Bus Tie Breaker, but operators will have to jumper out an interlock to make this happen.

D) Correct. To restore power to Division 2 ESF Bus 142Y, the cross ties will need to be use from Unit 2. Normally, this will prevent closing the tie breaker between 142Y and 142X, so a jumper will need to be installed with Shift Manager permission per LOA-AP-101.

Technical References: LOA-AP-101, Unit 1 AC Power System Abnormal, Rev 59; AP-3, Rev 4; 120 Plant Air Systems, Rev 12; LOA-IA-101, Loss of Instrument/Service Air, Rev 14

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 120.00.16 - Given various plant conditions, predict the Plant Air System response to a loss of the major power supplies

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

Question History: Last NRC Exam NA

Question Cognitive Level:
Memory or Fundamental Knowledge __
Comprehension or Analysis _3

10 CFR Part 55 Content: CFR 41.7, 45.6

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295021 AK1.01	
	Importance Rating	3.6	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to loss of shutdown cooling: decay heat

Proposed Question: **Question #9**

Unit 1 is in Mode 3 with RHR in Shutdown Cooling mode with the Main Condenser available when the following occurs:

- PCIS Group 6 lights are LIT on panel 1H13-P609
- 1C34-R605, WIDE RANGE RX PRESS indicates 155 psig and RISING
- The crew enters LOA-RH-101, “Unit 1 RHR Abnormal”

In LOA-RH-101, which control manipulation is performed FIRST to restore a method of decay heat removal?

A. Open 1E12-F008, RHR SHTDN CLG SUCT OUTBOARD ISOL VLV

B. Change pressure set to OPEN bypass valves

C. Bypass Div 1 SDC suction flow and high RPV pressure

D. Reset PCIS Group 6

Proposed Answer: B

Explanation: This meets the K/A statement because a loss of shutdown cooling on a valid Group 6 isolation will result in a loss of decay heat removal. The operational implications on a loss of shutdown cooling is to take actions to restore decay heat removal.

A) Incorrect. Plausible because when restoring decay heat removal IAW LOA-RH-101 Attachment D, the first valve that is opened is 1E12-F008. Decay heat removal is lost due to the valid PCIS Group 6 isolation from reactor pressure being greater than 135 psig. Both the inboard and outboard RHR shutdown cooling suction valves will close. In order to restore shutdown cooling flow, the reactor pressure must be lowered to less than 135 psig in order to reset the PCIS Group 6 isolation. Once Group 6 is reset, then 1E12-F008 can be opened.

B) Correct. With reactor pressure above 135 psig, this is a valid PCIS Group 6 isolation. This will cause the inboard and outboard RHR shutdown cooling suction valves to close, which will then trip the running RHR Pump. Before shutdown cooling flow can be restored and before the suction valves can be re-opened, reactor pressure must be below 135 pisg in order to reset PCIS Group 6. LOA-RH-101 step B.1.5 directs the crew to restore reactor pressure below 135 psig using the bypass valves.

C) Incorrect. Plausible if the examinee does not realize the given conditions indicate a valid PCIS Group 6 isolation has occurred. This step is directed in LOA-RH-101 if the Group 6 isolation is not valid. The condition of the reactor pressure greater than 135 psig indicates that this is a valid Group 6 isolation, therefore bypassing a valid PCIS signal would not be performed.

D) Incorrect. Plausible as this step is directed to restore shutdown cooling flow in LOA-RH-101 Attachment D prior to opening the inboard and outboard RHR shutdown cooling suction valves. However, a valid PCIS Group 6 is present due to reactor pressure greater than 135 psig. Therefore, the PCIS Group 6 cannot be reset until reactor pressure is lowered to less than 135 psig.

Technical Reference(s): 064 Residual Heat Removal Rev 17 (p. 2), 091 PCIS Rev 15 (p. 29), LOA-RH-101 Rev 22

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 064.05.12

Question Source:	Bank #	____	
	Modified Bank #	____	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	NA	

Question Cognitive Level:	Memory or Fundamental Knowledge	____
	Comprehension or Analysis	3

10 CFR Part 55 Content: 55.41 **8-10**

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295023 AA2.03	
	Importance Rating	3.3	

K/A Statement: Ability to determine and/or interpret the following as they apply to Refueling Accidents: Airborne Contamination Levels

Proposed Question: **Question #10**

During fuel movement activities on the refueling floor, a fuel assembly is accidentally dropped.

Per LOA-FH-001, “Irradiated Fuel Assembly Damage/ Removal of Rack,” the PRIMARY reason for evacuating personnel from the refuel floor is to minimize ____(1)____, and this condition will be made worse if ____(2)____.

A. 1) internal exposure from gaseous fission products
2) fuel pool LEVEL is at the LOWER limit

B. 1) external radiation exposure
2) fuel pool TEMPERATURE is at the UPPER limit

C. 1) internal exposure from gaseous fission products
2) fuel pool LEVEL is at the UPPER limit

D. 1) external radiation exposure
2) fuel pool TEMPERATURE is at the LOWER limit

Proposed Answer: A

Explanation:

A) Correct. If an irradiated fuel bundle is damaged, the primary concern is the release of gaseous fission products resulting in rising airborne contamination levels, and the condition is made worse with low fuel pool level because more of the iodine coming out of solution will make it to the surface of the fuel pool raising airborne contamination levels.

B) Incorrect. Plausible because high fuel pool temperature could lead to a rise in airborne contamination levels caused by increased evaporation from the fuel pool.

C) Incorrect. Plausible because high fuel pool level could result in the fuel pool overflowing into the ducts of the Reactor Building ventilation system leading to the spread of contamination.

D) Incorrect. Plausible because fuel pool temperatures at the lower limit would increase the solubility of fission product gasses allowing the gasses to accumulate in the fuel pool water raising external exposure for those nearest to the fuel pool.

Technical Reference(s): Lesson Plan 118 Rev 13 (P. 16), 1H13-P601-F205 Rev 5, UFSAR 15.7.4 (P. 16a), Lesson Plan 095 Rev 11 (P. 1)

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 118.00.05g; Given the following VR System key parameters, recall the physical location of indicators, local and remote, and describe the sensor location in the system flowpath while operating the system, during abnormal conditions, or on an exam in accordance with the student text: j. Fuel Pool Ventilation Exhaust Radiation Level

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

Question History: Last NRC Exam LaSalle 2016 ILT Exam (Q78)

Question Cognitive Level: Memory or Fundamental Knowledge 2
Comprehension or Analysis ___

10 CFR Part 55 Content: 41.1 / 43.5 / 45.13

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295024 G2.2.42	
	Importance Rating	3.9	

K/A Statement: Ability to recognize system parameters that are entry-level conditions for Technical Specifications: High Drywell Pressure

Proposed Question: **Question #11**

The following conditions exist on Unit 1:

- Suppression pool temperature is 100°F and STABLE
- Suppression pool level is +2.5 inches and STABLE
- Drywell temperature is 120°F and STABLE
- Drywell pressure is 0.8 psig and STABLE

Which of the following requires entry into Technical Specifications?

A. Suppression pool temperature

B. Suppression pool level

C. Drywell temperature

D. Drywell pressure

Proposed Answer: D

Explanation:

This question gives a set of conditions and plant parameters, and the examinee must have the ability to recognize which parameters meet entry conditions into the Tech Specs.

A) Incorrect. Plausible because suppression pool temperature is greater than normal/expected at 105 degrees, and is right at the TS limit. However TS 3.6.2.1 requires the suppression pool temperature to be less than or equal to 105 degrees, therefore there is no required action for suppression pool temperature.
B) Incorrect. Plausible the suppression pool level is greater than normal/expected at +3.0 inches, and is right at the TS limit. However, TS 3.6.2.2 requires the suppression pool level to be less than/equal to +3.0 inches and greater than/equal to -4.5 inches, therefore there is no required action for suppression pool level.
C) Incorrect. Plausible as the drywell temperature is greater than normal/expected at 120 degrees. However, TS 3.6.1.5 requires drywell temperature to be less than/equal to 135 degrees. Due to instrumentation error, the surveillance requires the temperature indication at 2PM05J to be less than/equal to 123 degrees. The given condition is that drywell temperature indicates 120 degrees, so there is no required action for drywell temperature.
D) Correct. Drywell pressure is greater than +0.75 psig, which places Unit 2 in LCO 3.6.1.4.A. The required action of the crew is to restore drywell pressure to less than or equal to 0.75 psig and greater than/equal to -0.5 psig in one hour.

Technical Reference(s): 3.6.1.4 Drywell and Suppression Chamber Pressure, 3.6.1.5 Drywell Air Temperature, 3.6.2.1 Suppression Pool Average Temperature, 3.6.2.2 Suppression Pool Water Level

Proposed references to be provided to applicants during examination: N/A

Learning Objective:

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

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge 3
Comprehension or Analysis ___

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

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295025 EK1.06	
	Importance Rating	3.5	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to High Reactor Pressure: Pressure effects on reactor water level.

Proposed Question: **Question 12**

Unit 2 was operating at 100% RTP:

- A Reactor High pressure SCRAM occurs due to a Turbine Trip
- Two SRVs OPEN

When the SRVs open, RPV level will **INITIALLY ________**.

A. Remain the same

B. Rise

C. Lower

D. Oscillate

Proposed Answer: B

Explanation:

A) Incorrect. The opening of the two SRVs will cause a rapid drop in RPV pressure. No other failures have been given in the stem so the main turbine bypass valves will still be available and controlling RVP pressure. They will close as the pressure drops but the steam leaving will still exceed the steam generated by decay heat and pressure will drop. Plausible if the candidate assumes that the main turbine bypass valves will close and maintain RPV pressure stable. The drop in pressure will cause RPV level to initially rise.

B) Correct. The opening of the two SRVs will cause a rapid drop in RPV pressure. No other failures have been given in the stem so the main turbine bypass valves will still be available and controlling RVP pressure. They will close as the pressure drops but the steam leaving will still exceed the steam generated by decay heat and pressure will drop. As pressure drops, voids will increase in the core region, resulting in a large differential pressure which much be overcome by static head of water in the downcomer region of the core. Therefore, level will initially rise.

C) Incorrect. Plausible if the applicant believes that the SRVs opening represents a loss in RCS inventory which will be indicated by a lowering RPV level.

D) Incorrect. Plausible if the applicant determines that the lowering pressure from the SRVs opening will create additional voiding in the core which will result in flow oscillations associated with unpredictable void formation and collapsing in the core.

Technical Reference(s): LGA-001, Revision 18 and associated Lesson Plan Revision 25; Lesson Plan 070 Main Steam, Revision 10;

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 410.00.01

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

Question History: Last NRC Exam NA –

Question Cognitive Level: Memory or Fundamental Knowledge _-_
Comprehension or Analysis 2

10 CFR Part 55 Content: 41.8, 41.9, 41.10

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

K/A Statement: Knowledge of the interrelations between suppression pool high water temperature and the following: Suppression pool spray

Proposed Question: **Question #13**

Unit 1 experienced a design basis LOCA and the 'A' RHR Heat Exchanger has just been placed in service to support Suppression Pool Cooling IAW LGA-RH-103, "Unit 1 A/B RHR Operations in the LGAs/LSAMGs."

- Suppression pool temperature is 106ºF and RISING
- Suppression pool level is -8 inches and LOWERING
- Suppression chamber pressure is 20 psig and RISING
- Reactor pressure is 540 psig and LOWERING at 10 psig per minute
- The INJECTION OVERRIDE Switches are in NORMAL

FIVE minutes later, which actions must the operators perform NEXT in order to:

(1) Provide additional cooling to the suppression pool, AND
(2) Initiate suppression chamber spray

A. (1) Throttle closed the '1A' RHR Heat Exchanger Outlet Valve
(2) Open 'A' Suppression Chamber Spray Valve

B. (1) Throttle closed the '1A' RHR Heat Exchanger Bypass Valve
(2) Open 'A' Suppression Chamber Spray Valve

C. (1) Throttle closed the '1A' RHR Heat Exchanger Outlet Valve
(2) Close 'A' LPCI Injection Valve

D. (1) Throttle closed the '1A' RHR Heat Exchanger Bypass Valve
(2) Close 'A' LPCI Injection Valve

Proposed Answer: __D__

Explanation:

With high suppression pool water temperature, additional cooling is provided by closing the Hx bypass valve. Suppression chamber spray can also be initiated. During a design basis LOCA when suppression pool cooling is aligned, the suppression chamber spray valve can only be opened to initiate flow as long as the LPCI injection valve is closed.

A) Incorrect. Plausible because LGA-RH-103 directs the operators to throttle RHR Hx valves, however the outlet must be opened and the bypass closed. Suppression chamber spray is initiated once the ‘A’ Suppression Chamber Spray Valve is open. With a design basis LOCA, the LPCI Injection Valve is open due to low vessel level and/or high drywell pressure. There is an interlock where the Suppression Chamber Spray Valve will not open if the LPCI Injection Valve is open. The given information states the ‘A’ RHR Hx has just been placed in service. IAW with LGA-RH-103, if it is desired to initiate suppression chamber flow, then the LPCI Injection Valve must be fully closed before the Suppression Chamber Spray Valve can be opened.
B) Incorrect. Plausible as the RHR Hx bypass valve is throttled closed to provide additional cooling. With a design basis LOCA, the RPV is depressurized and therefore the LPCI Injection Valve is open. In order to initiate suppression chamber spray, the LPCI Injection valve must be fully closed.
C) Incorrect. The RHR Hx outlet must be opened and the bypass must be closed. The second part of the distractor is correct.
D) Correct. Closing the RHR Hx bypass will provide additional cooling to the suppression pool. With the RPV depressurized less than 500 psig, the LPCI Injection Valve must be fully closed before the suppression chamber spray valve can be opened to provide suppression chamber spray.

Technical Reference(s): 064 Residual Heat Removal Rev 17 (p. 23), RH-1, RH-2, LGA-RH-103 Rev 13

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 064.00.03d, 064.00.04e

Question Source: Bank # _____
Modified Bank # _____ (Note changes or attach parent)
New __X__

Question History: Last NRC Exam _NA_

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis __3_

10 CFR Part 55 Content: 55.41 __7_
55.43 ____

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295028 EK3.02	
	Importance Rating	3.5	

K/A Statement: Knowledge of the reasons for the following responses as they apply to high drywell temperature: RPV Flooding

Proposed Question: **Question 14**

Entry into LGA-005, "RPV Flooding" during an accident is required for ______________.

A. Drywell temperature exceeding the saturation curve of LGA-001, Figure J resulting in unknown RPV water level

B. High Suppression Pool temperature which cannot be maintained below the Heat Capacity Temperature Limit Curve

C. Low Suppression Pool water level below -12 feet

D. RPV water level not able to be restored and maintained above TAF

Proposed Answer: A

Explanation:

A) Correct. The entry criteria for LGA-005 is given in LGA-001, 004, 006, 010 as RPV water level unknown. In LGA-001 Detail I, Figure J or K is used to determine RPV level instrument reliability.

B) Incorrect. Plausible as an abnormally high suppression pool temperature not able to be maintained below the HCTL Curve is a transition point from LGA-003 to RPV Blowdown LGA-004/6.

C) Incorrect. Plausible as this low suppression pool level is a transition point from LGA-003 to LGA-004/6 RPV Blowdown. A correction to this condition would be to add water to the suppression pool which could be confused with RPV Flood.

D) Incorrect. Plausible as this is a transition point from LGA-001 to LGA-004 RPV Blowdown and would be corrected by adding inventory to the RCS.

Technical Reference(s): LGA-001, RPV Control, Rev 18; LGA-005, RPV Flooding, Rev 17; Lesson Plan for LGA-001, RPV Control, Rev 25

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 414.00.02, (from LGA-001) Given plant conditions and LGA entry, utilize Detail I, RPV Water Level Instruments, to determine which RPV water level indicators may be used while operating the plant or on an exam, IAW the LGAs.

Question Source:

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

Question History: Last NRC Exam NA –

Question Cognitive Level:

	Memory or Fundamental Knowledge	2
	Comprehension or Analysis	___

10 CFR Part 55 Content: 55.41.5 and 41.6

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295030 EA1.03	
	Importance Rating	3.4	

K/A Statement: Ability to operate and/or monitor the following as they apply to low suppression pool water level: HPCS

Proposed Question: **Question #15**

A seismic event created a LOCA and a leak in the suppression chamber on Unit 1. Suppression pool level is -4.5 inches and rapidly lowering.

Which of the following is a concern for prolonged operation of the HPCS Pump?

A. Damage due to cavitation

B. Trip on low suppression pool level

C. Damage due to pump run-out

D. Overfilling the RPV

Proposed Answer: A

Explanation:

The suppression pool is the suction source for the HPCS pump. In a severe case where suppression pool lowers to -18ft, the HPCS may cavitate, and operators should be monitoring pump parameters for cavitation.

A) Correct. Prolonged operation of HPCS with a leak in the suppression pool will result in loss of NPSH to HPCS. Lesson Plan 061 HPCS System states operation of HPCS with a low suppression pool level may cause cavitation and pump damage.
B) Incorrect. Plausible as low suppression pool level may cause low suction pressure. Several pumps have a low suction pressure trip setpoint. However, the HPCS pump does not have an automatic low suction pressure trip.
C) Incorrect. Plausible as damage due to pump run-out is a concern because reactor pressure is lowering due to the LOCA. Therefore, the d/p between the RPV and HPCI pump suction will lower as the LOCA progresses and HPCS pump flow will increase. Lesson Plan 061 HPCS System states there is an injection line flow orifice that will restrict flow in order to prevent pump run-out as the d/p lowers.
D) Incorrect. Plausible as the RPV level will rise during HPCS pump operation, as well as the operation of all make-up sources. The HPCS pump will continue to run until RPV level reaches the level 8 setpoint, then it will secure to prevent overfilling the RPV.

Technical Reference(s): HP-1, 061 HPCS System Rev 9 (p. 19), LGA-003 Rev 18

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 061.00.16a

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

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge 2
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 7
55.43 ____

Comments:

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295031 EA2.04	
Importance Rating	4.6	

K/A Statement: Ability to determine and/or interpret the following as they apply to reactor low water level: Adequate core cooling

Proposed Question: **Question 16**

Unit 1 was operating at 100% RTP when a Design Basis Loss of Coolant Accident (LOCA) occurs:

- Reactor scram
- LPCI pumps are NOT available
- RPV level is -200 inches on FZ indication and RISING
- RPV pressure is 18 psig
- HPCS is injecting at 3850 gpm
- LPCS is injecting at 4650 gpm

What is the CURRENT state of adequate core cooling in accordance with LGA-001, “RPV Control,” Detail AC?

AC — Adequate Core Cooling for LGA-001

Adequate Core Cooling exists if one of the following is met:

☐ 1. Water level in the RPV ≥ –150 in. on WR.

OR

☐ 2. Water level in the RPV ≥ –183 in. on FZ.

OR

☐ 3. Adequate Core Cooling conditions exist for DBA LOCA:

- Water level in the RPV ≥ –210 in. on FZ (Top of Jet Pump risers).

AND

- At least one core spray system injecting onto the core at ≥ 6250 gpm (HPCS or LPCS).

AND

- RPV pressure ≤ 20 psig.

A. Adequate core cooling exists because level is above 2/3 core height and RISING with multiple injection sources available

B. Adequate core cooling does NOT exist because RHR/LPCI and LPCS must both be available to inject during a design basis LOCA

C. Adequate core cooling exists because RPV water level is ≥ -210 inches on FZ, total core spray cooling flow rate exceeds 6250 gpm, and RPV pressure is ≤ 20 psig

D. Adequate core cooling does NOT exist because neither HPCS nor LPCS is individually achieving an adequate spray pattern and flow rate to each fuel assembly

Proposed Answer: D

Explanation:

A) Incorrect. Plausible as the applicant may assess that with a RISING reactor water level above 2/3 core height that spray cooling is effective in adequately removing decay heat.

B) Incorrect. Plausible as the applicant may assess that with no LPCI available for injection and the reactor depressurized that adequate core cooling cannot be achieved.

C) Incorrect. Plausible because reactor water level and pressure meet the criteria for adequate core cooling per Detail AC and total spray flow exceeds 6250 gpm. For spray flow to be effective, either LPCS or HPCS must individually exceed 6250 gpm.

D) Correct. To meet the requirements of adequate core cooling during a DB LOCA if the core is not submerged, then level has to be ≥ -210 in FZ; HPCS ***or*** LPCS has to be injecting ≥ 6250 gpm to achieve an adequate spray pattern ensuring adequate flow to each fuel assembly; and RPV pressure is ≤ 20 psig. Total core spray flow is not an acceptable measurement of cooling effectiveness.

Technical Reference(s): Lesson Plan for LGA-001, (p. 37) Rev 25; LGA-001, RPV Control Rev 18

Proposed references to be provided to applicants during examination: LGA-001, Detail AC

Learning Objective: 400.00.08; Given plant conditions, LGA entry and RPV injection system status and level trend, determine if core cooling will be assured, while operating the plant or on an exam, IAW the LGA procedures.

Question Source:	Bank #	___	
	Modified Bank #	___ (Note changes or attach parent)	
	New	_X_	
Question History:	Last NRC Exam	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge		__
	Comprehension or Analysis		_2
10 CFR Part 55 Content:	41.10, 43.5, 45.13		

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295037 G2.1.31	
	Importance Rating	4.6	

K/A Statement: Scram Condition Present and Reactor Power Above APRM Downscale or Unknown: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Proposed Question: **Question #17**

Unit 1 has experienced a high power ATWS with the following plant conditions observed:

- 1H13-P603-B512, 1B RPS MG SET TROUBLE is LIT
- Reactor Power Module on the SPDS indicates CYAN

Which of the following indications is available to determine Reactor Power?

A. IRM/APRM/RBM Recorder (1C51-R603C) IRM E,G/ APRM E/ RBM A CHANNEL

B. Power to Flow Map Display

C. IRM/APRM/RBM Recorder (1C51-R603D) IRM F,H/ APRM F/ RBM B CHANNEL

D. Safety Parameter Display System (SPDS)

Proposed Answer: ___A__

Explanation:

Unit 1 has an ATWS and has also lost RPS Bus 'B.' The SPDS indicating CYAN also indicates a loss of power indication on the plant computer. The only indication to reactor power is from indications powered from RPS Bus 'A.'

A) Correct. APRM 'E' is powered from RPS Bus 'A'. Under these conditions, RPS Bus 'A' remains powered, and the IRM/APRM/RBM (1C51-R603C) recorder located on panel 1H13-P603 can be used to determine Reactor Power as it displays APRM 'E' indication.
B) Incorrect. Plausible as the power to flow map gives an indication of location on the map, which can be used to determine core flow and reactor power. With the inputs to the Reactor Power Module being bad (indicating CYAN), the power to flow map display cannot be used to determine Reactor Power.
C) Incorrect. Plausible as the APRM recorders can be used to determine Reactor Power. However, the APRM 'F' which is displayed on (1C51-R603D) recorder is powered from RPS Bus 'B' which loses power based on the provided conditions. Therefore, APRM 'F' cannot be used to determine Reactor Power.
D) Incorrect. Plausible because the Plant Process Computer will display reactor power via the Reactor Power Module on SPDS. According to Lesson Plan 050 Process Computer, if the SPDS module indicates CYAN, that means the input is bad and therefore cannot be used to determine Reactor Power.

Technical Reference(s): 044 APRM Rev 8 (p. 21), 050 Process Computer Rev 5 (p. 15), LOA-NR-101 Rev 20, MCR Images 118 and 120

Proposed references to be provided to applicants during examination: N/A

Learning Objective:

Question Source:	Bank #	_____	
	Modified Bank #	_____	(Note changes or attach parent)
	New	__X__	
Question History:	Last NRC Exam	_NA_	
Question Cognitive Level:	Memory or Fundamental Knowledge	____	
	Comprehension or Analysis	__3_	
10 CFR Part 55 Content:	55.41	**_8-10_**	
	55.43	____	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295038 EA2.04	
	Importance Rating	4.1	

K/A Statement: Ability to determine and/or interpret the following as they apply to high off-site release rate: Source of off-site release.

Proposed Question: **Question 18**

A transient has occurred causing Unit 2 reactor fuel failure.

Over the next 10 minutes:

- The MSIVs and Main Steam Line Drains were CLOSED by the crew due to elevated steam line radiation
- PCIS Groups 2, 4, 7, and 10 have isolated
- VG WRGM indicates 7E6 µCi/sec

What is the MOST LIKELY source of the rising release rate?

Leakage FROM the…

A. Primary Containment TO the Secondary Containment

B. Reactor Coolant Pressure Boundary TO the Offgas System

C. Secondary Containment TO the Auxiliary Building

D. Reactor Coolant Pressure Boundary TO the Primary Containment Cooling System

Proposed Answer: ___A___

Explanation:

A) Correct. The isolations that occurred are all actuated by elevated drywell pressure. Leakage from the primary containment to the secondary containment (reactor building) would be expected. In order to be indicated on the VG WRGM, the leakage must be discharged into the reactor building.

B) Incorrect. Plausible as a release of radioactivity to the offgas system would travel out of the station vent stack.

C) Incorrect. Plausible as the applicant could assess that the radioactive release has escaped the secondary containment and as such is indicated by the WRGM data.

D) Incorrect. Plausible has a release of radioactive material from the RCS to primary containment cooling equipment is a primary containment bypass path which would be a means for a release pathway into the secondary containment.

Technical Reference(s): 095, Standby Gas Treatment System (VG), Rev 11

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 095.00.02; Recall the Standby Gas Treatment flowpath during system operations or by drawing a one-line diagram on an exam in accordance with Figure 095-01.

Question Source:	Bank #	X
	Modified Bank #	_ (Note changes or attach parent)
	New	
Question History:	Last NRC Exam	LaSalle 2000 NRC ILE (Q33)
Question Cognitive Level:	Memory or Fundamental Knowledge	_
	Comprehension or Analysis	3
10 CFR Part 55 Content:	41.10 / 43.5 / 45.13	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	600000 G2.1.20	
	Importance Rating	4.6	

K/A Statement: Plant Fire Onsite: Ability to interpret and execute procedure steps.

Proposed Question: **Question #19**

The following has occurred on Unit 2 at 0800:

- A fire in the Turbine Building
- Fire Header Pressure reached a low of 122 psig and then stabilized

At 0830, the following is observed:

- The fire is extinguished
- Fire Header Pressure is 160 psig and STABLE

IAW LOP-FP-02, "Fire Pump Diesel Startup and Shutdown," the crew will ___(1)___ from the ___(2)___.

A. (1) Stop the 0A Diesel Fire Pump ONLY
(2) Main Control Room

B. (1) Stop both Diesel Fire Pumps
(2) Main Control Room

C. (1) Stop the 0A Diesel Fire Pump ONLY
(2) Lake Screen House

D. (1) Stop both Diesel Fire Pumps
(2) Lake Screen House

Proposed Answer: __C__

Explanation:

In this situation, a fire has occurred on Unit 2. The demand from the fire protection system only requires one diesel fire pump. The 0A Diesel Fire Pump auto starts at 124 psig fire header pressure and the 0B Diesel Fire Pump auto starts at 120 psig. During the fire, the fire header pressure did not lower to the setpoint of the second fire pump starting. An operator must secure a running Diesel Fire Pump locally in the Lake Screen House.

A) Incorrect. Plausible because only the 0A Diesel Fire Pump will be running. In the MCR, the fire pump can be manually started, but it cannot be secured, that must be done locally in the Lake Screen House.
B) Incorrect. Plausible as fire suppression will cause fire header pressure to lower until the Diesel Fire Pumps auto start to maintain pressure. In this case, the pressure did not lower to the point of the second Diesel Fire Pump starting. In the MCR, the fire pump can be manually started, but it cannot be secured, that must be done locally in the Lake Screen House.
C) Correct. Fire suppression will cause fire header pressure to lower, and the 0A Diesel Fire Pump will start at 124 psig. Since the header pressure was maintained at 122 psig, the 0B Diesel Fire Pump will not start. A running Diesel Fire Pump must be secured from the Lake Screen House.
D) Incorrect. Plausible as fire suppression will cause fire header pressure to lower until the Diesel Fire Pumps auto start to maintain pressure. In this case, the pressure did not lower to the point of the second Diesel Fire Pump starting. The second part is correct, a running Diesel Fire Pump must be secured from the Lake Screen House.

Technical Reference(s): 125 Fire Protection Rev 12 (p. 7), LOP-FP-02 Rev 26, LOA-FP-201 Rev 41

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 125.00.06a1

Question Source:
Bank # _____
Modified Bank # _____ (Note changes or attach parent)
New __X__

Question History: Last NRC Exam NA

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __2__

10 CFR Part 55 Content:
55.41 __**10**__
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	700000 AK1.02	
	Importance Rating	3.3	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to generator voltage and electric grid disturbances: Over-excitation

Proposed Question: **Question 20**

Unit 1 is at 100% RTP.

A grid disturbance on a 345 KV line results in the following indication:

- 1PM01J-A112, U1 GEN EXCITATION TROUBLE, is LIT (R0350, Generator Max Excitation Alarm)

If the grid disturbance and alarming conditions last for ONE MINUTE, what is the expected plant response and associated required operator action?

A. The Generator Field Breaker trips resulting in a turbine trip and reactor scram; Perform the actions of LOA-TG-101, “Unit 1 Turbine Generator” and LGP 3-2, “Reactor SCRAM”

B. Generator MVAR output will RISE and the Voltage Regulator will shift to MANUAL; Control alternator voltage and current in MANUAL

C. Generator MWe output will RISE; Lower reactor power to return to the pre-transient condition

D. Generator MVAR output will RISE and the Voltage Regulator will remain in AUTO; RESET the alarming condition by cycling the Voltage Regulator to MANUAL then back to AUTO

Proposed Answer: A

Explanation:

A) Correct. Annunciator Response Procedure LOR-1PM01J-A112, step B.4, states that if generator maximum excitation exists for greater than 10 seconds, the field breaker will trip. Operators will have to respond to a turbine trip and with the unit at full power a reactor scram.

B) Incorrect. Plausible as the maximum excitation condition will cause reactive power output to increase as a greater magnetic field flux will occur in the generator and the Voltage Regulator will transfer to MANUAL when the condition lasts for greater than 5 seconds.

C) Incorrect. Plausible as the applicant may confuse the maximum excitation condition as affecting MWe.

D) Incorrect. Plausible as MVARs increase, but if the condition lasts for greater than 5 seconds the Voltage Regulator will on its own shift to MANUAL.

Technical Reference(s): LOR-1PM01J-A112, Rev 4, U1 GEN EXCITATION TROUBLE; 008, Main Generator and Generator Excitation Rev 10; TG-3, Main Generator Rev 2

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 008.00.05j; Recall the function, theory of operation, interlocks, trips, and characteristics of the following Main Generator System components and relate these items to overall system operation while operating the system or on an exam in accordance with the student text…j. Exciter

Question Source:	Bank #	___	
	Modified Bank #	_	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	3	
	Comprehension or Analysis	-	
10 CFR Part 55 Content:	41.4 / 41.5 / 41.7 / 41.10 / 45.8		

Comments: Oyster Creek LER - On July 12, 2009 with the Unit at 100% power in the "Power Operation" mode, a severe electrical storm resulted in multiple lightning strikes on an interconnected 34.5 kV offsite transmission line. These lightning strikes in conjunction with a failure of a line breaker to open caused grid disturbances, a main generator trip on over-excitation, an automatic reactor scram due to the load rejection, and a loss of offsite power to the Startup Transformers.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295007 AA2.02	
	Importance Rating	4.1	

K/A Statement: Ability to determine and/or interpret the following as they apply to high reactor pressure: Reactor power.

Proposed Question: **Question #21**

Unit 2 was at 100% RTP when the Inboard MSIVs on Main Steam Lines ‘A’ and ‘C’ simultaneously failed CLOSED. Which of the following describes the expected plant response?

A. RPS will cause a scram DIRECTLY from the MSIV Closure

B. Reactor power will rise until the reactor scrams

C. TCVs and TBVs will OPEN and control reactor pressure

D. Main turbine will trip on overspeed

Proposed Answer: B

Explanation:

Two MSIVs closing at 100% RTP failing closed will cause reactor pressure to increase, which causes voids to collapse. When the voids collapse, reactor power will increase and the reactor will scram on high flux. With two MSIVs closing on MSL A and C, only RPS B will trip on MSIV Closure. The TCVs and TBVs will open, but the pressure transient will cause the reactor to scram before the TCVs and TBVs can control reactor pressure.

A) Incorrect. Plausible because RPS will scram the reactor on MSIV Closure if one MSIV closes on three different Main Steam Lines. With one on MSL A and C, only RPS B will trip, causing a half scram.
B) Correct. When the two MSIVs close, reactor pressure will increase, causing voids to collapse. Reactor power will increase and the reactor will scram on high flux.
C) Incorrect. Plausible because the TCVs and TBVs will open. However, at 100% with 2 MSIVs closing, the pressure transient will cause reactor power to rise until the reactor scrams on high flux before the TCVs and TBVs can control reactor pressure.
D) Incorrect. Plausible because in other situations, an increase in reactor pressure with all the TCVs and TBVs open would cause steam flow to increase, and therefore cause the turbine to trip on overspeed. In this case, the increase in pressure is due to two MSIVs closing, and steam flow to the turbine will not cause the turbine to overspeed.

Technical Reference(s): 049 Reactor Protection System Rev 11 (pp. 24, 27, and 45), 070 Main Steam Rev 10 (pp. 13-15), LOR-1H13-P603-B105 Rev 4, MS-1

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 049.00.10

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

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis 3

10 CFR Part 55 Content: 55.41 **10**
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295012 G2.4.20	
	Importance Rating	3.8	

K/A Statement: High Drywell Temperature: Knowledge of the operational implications of EOP warnings, cautions, and notes.

Proposed Question: **Question 22**

What is the operational implications of the note in LGA-003, when directed to "Start all available drywell cooling (LGA-VP-01)" that states, "Cannot be used in LOCA if Drywell temperature goes above 212 degrees Fahrenheit"?

A. Use of Primary Containment Cooling (VP) in these conditions may cause water hammer and VP piping damage, creating a leak in the Drywell

B. Use of VP in these conditions is beyond system design cooling

C. Use of VP in these conditions will likely cause an overload trip of the Drywell Cooling Supply Fans

D. Use of VP in these conditions will cause VP to automatically isolate

Proposed Answer: ___A___

Explanation:

A) Correct. This Note is applicable for High Drywell Temperature conditions > 135ºF in LGA-003 during LOCA conditions as water hammer could cause pipe damage in containment which could become a containment bypass.

B) Incorrect. Plausible as VP is non-ESF system and therefore is not expected to be operable during design basis events.

C) Incorrect. Plausible as a LOCA will result in a high humidity condition in the drywell which will add resistance to cooling fan operation.

D) Incorrect. Plausible as the primary purpose of the VP system is to maintain a suitable environment inside the drywell for equipment operation and longevity during non-LOCA operation conditions.

Technical Reference(s): LGA-003, Primary Containment Control, Rev 18; LGA-VP-01, Primary Containment Temperature Reduction, Rev 8; VP-1, Containment Vent & Cooling, Rev 1

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 400.00.18; Given LGA entry, recall the general rules for LGA flowchart utilization while operating the plant or on an exam, IAW the student text.

Question Source:	Bank #	___
	Modified Bank #	___ (Note changes or attach parent)
	New	_X_
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory or Fundamental Knowledge	___
	Comprehension or Analysis	_3_
10 CFR Part 55 Content:	41.10 / 43.5 / 45.13	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295014 AK1.05	
	Importance Rating	3.7	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to inadvertent reactivity addition: Fuel thermal limits.

Proposed Question: **Question #23**

Unit 1 was at 90% RTP and ascending to 100% RTP following a refueling outage when the following occurred:

- 1H13-P603-A504, CRD DRIFT, is LIT
- Crew entered LOA-RD-101, "Control Rod Drive Abnormal"
- One control rod DRIFTED OUT to position 08 and is stuck

In accordance with LOA-RD-101, the crew will ___(1)___ in order to ___(2)___.

A. (1) lower power by 100 MWe using RR FCV's
(2) prevent thermal hydraulic instabilities

B. (1) lower power by 100 MWe using RR FCV's
(2) prevent violating fuel design limits

C. (1) manually scram the reactor
(2) prevent thermal hydraulic instabilities

D. (1) manually scram the reactor
(2) prevent violating fuel design limits

Proposed Answer: B

Explanation:

A drifting control rod will significantly reduce power around the control rod due to a negative reactivity addition and result in nodal power increases in other regions of the core due to the redistribution of core flow and change of flux shape which causes a positive reactivity addition. In LOA-RD-101, Discussion C.3.2 states when a rod cannot be inserted to at least position 04, a 100 MWe power reduction is directed to protect the fuel cladding

A) Incorrect. Plausible as the first part of the distractor is correct. The second part is also plausible because reducing power using RR may result in thermal hydraulic instability by entering Region 2 of the power to flow map. A drift of a control rod is a change in reactivity and will not cause thermal hydraulic instabilities.
B) Correct. The conditions state that the rod will not insert past position 08. IAW LOA-RD-101, when a rod cannot be inserted to at least position 04, power is to be reduce by 100 MWe in order to protect the fuel cladding.
C) Incorrect. Plausible because a reactor scram is required for drifting control rods, but there must be more than 1 rod moving at the same time, or 3 or more rods scrammed or fully inserted. These conditions are not met, so a manual scram is not required. The second part is also plausible because reducing power using RR may result in thermal hydraulic instability by entering Region 2 of the power to flow map. A drift of a control rod is a change in reactivity and will not cause thermal hydraulic instabilities.
D) Incorrect. Plausible because a reactor scram is required for drifting control rods, but there must be more than 1 rod moving at the same time, or 3 or more rods scrammed or fully inserted. These conditions are not met, so a manual scram is not required. The second part is correct.

Technical Reference(s): B 3.1.2 Reactivity Anomalies, B 2.1.1 Reactor Core SLs, 021 Nuclear Fuel, LOA-RD-101

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 021.00.016g

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

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis 2

10 CFR Part 55 Content: 55.41 **8-10**
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295020 AK2.01	
	Importance Rating	3.6	

K/A Statement: Knowledge of the interrelations between inadvertent containment isolation and the following: Main Steam System

Proposed Question: **Question 24**

Unit 1 is at 100% RTP

- 1PM06J-A206, RB 1A VENT SPLY FAN AUTO TRIP, is LIT

(1) What is the concern if NO operator action is taken?
(2) What is done FIRST per LOR-1PM06J-A206?

A. (1) Main Steam Tunnel Differential Temperature may cause an MSIV Closure
(2) IMMEDIATELY start the standby VR Supply Fan

B. (1) Reactor Building Differential Pressure may cause leakage out of the Secondary Containment
(2) IMMEDIATELY start the standby VR Supply Fan

C. (1) Main Steam Tunnel Differential Temperature may cause an MSIV Closure
(2) Take the MSL Pipe Tunnel Differential Temperature Bypass keylock switches to BYPASS

D. (1) Reactor Building Differential Pressure may cause leakage out of the Secondary Containment
(2) Take the MSL Pipe Tunnel Differential Temperature Bypass keylock switches to BYPASS

Proposed Answer: A

Explanation:

A) Correct. Per the CAUTION in LOR-1PM06J-A206, with one supply fan running an MSIV closure on differential temperature could occur due to a leak detection isolation. Starting the standby VR supply fan addresses this concern.

B) Incorrect. Plausible as an imbalance between the number of running VR supply and exhaust fans will have an effect on RB differential pressure. In this case, the loss of a supply fan will result in a more negative DP in the RB, therefore, there is no concern of a radioactivity exiting the secondary containment. The second part of the answer is correct.

C) Incorrect. Plausible as part 1 of the answer is correct. Part 2 would prevent the MSIV closure on the current plant conditions, but it will prevent a required plant response if subsequent conditions require it and it does not address the abnormal VR lineup.

D) Incorrect. Plausible as an imbalance between the number of running VR supply and exhaust fans will have an effect on RB differential pressure. In this case, the loss of a supply fan will result in a more negative DP in the RB, therefore, there is no concern of a radioactivity exiting the secondary containment. Part 2 would prevent the MSIV closure on the current plant conditions, but it will prevent a required plant response if subsequent conditions require it and it does not address the abnormal VR lineup.

Technical Reference(s): LOR-1PMO6J-A206, RB 1A VENT SPLY FAN AUTO TRIP, Rev 3; MS-2, Rev 3

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 070.00.18, Given various plant conditions, predict the response of the following supported systems to a loss of the Main Steam System while operating the system, or on an exam in accordance with student text: Primary Containment Isolation System

Question Source:	Bank #	___	
	Modified Bank #	___	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	3	
	Comprehension or Analysis	___	
10 CFR Part 55 Content:	41.7 / 45.8		

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295022 AK2.03	
	Importance Rating	3.4	

K/A Statement: Knowledge of the interrelations between loss of CRD pumps and the following: Accumulator pressures.

Proposed Question: **Question #25**

Unit 1 is at 100% RTP when the following occurred:

- 1H13-P603-A204, CRD CHARGING WTR PRESS LO is LIT
- CRD Cooling Wtr Flow on 1C11-R605 indicates 0 gpm
- CRD Drive Wtr Flow on 1C11-R605 indicates 0 gpm

Based on the above conditions, the scram accumulators are ______________.

A. NOT impacted

B. UNABLE to discharge if a scram is required

C. discharging to the charging water header

D. UNABLE to recharge following a scram

Proposed Answer: D

Explanation:

The given conditions indicate the running CRD Pump has tripped. As a result, Scram Accumulator water pressure will lower until the back-up CRD pump is manually started.

A) Incorrect. Plausible based on the given conditions, no HCU Accumulator alarms are present. Also, the accumulators will still be able to discharge on a scram, making it plausible that there is no impact. Without CRD header flow, the Accumulator Pressure is lowering which will impact control rod insertion times, so the scram accumulators are impacted.
B) Incorrect. Plausible as the reactor pressure may supplant accumulator water pressure (025 CRD Hydraulic) due to the loss of the CRD Pump. The accumulators will still discharge once reactor pressure lowers. The accumulators may become inoperable by not being able to insert control rods within specified time due to lowering pressure.
C) Incorrect. Plausible as the water pressure in the accumulator will be greater than the charging water header pressure. However, based on RD-1 and training material 025 CRD Hydraulic, a check valve prevents the accumulators from discharging.
D) Correct. Based on the given conditions, the running CRD Pump has trip, and the charging water header keeps the water in the accumulators pressurized. Without a running CRD Pump, accumulator pressure will not recharge following a scram once the HCU discharge to move control rods into the core.

Technical Reference(s): 024 CRD Mechanical Rev 8 (p. 29), RD-1, 3.1.5 Control Rod Scram Accumulators, B 3.1.5 Control Rod Scram Accumulators

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 025.00.05a, 025.00.18d

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

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis 2

10 CFR Part 55 Content:
55.41 7
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295029 EK3.03	
	Importance Rating	3.4	

K/A Statement: Knowledge of the reasons for the following responses as they apply to High Suppression Pool Water Level: Reactor SCRAM

Proposed Question: **Question 26**

What is the reason a scram is required if suppression pool level cannot be maintained BELOW LGA-003, “Primary Containment Control,” Figure S, SRV Tail Pipe Level Limit?

A. To ensure adequate margin for SRV operation to prevent SRV system damage and containment failure.

B. To shutdown the reactor before the containment pressure instruments become submerged.

C. To maintain suppression pool water level below 722 ft to ensure suppression chamber vacuum breakers remain above water.

D. To prepare for emergency depressurization as the Pressure Suppression Pressure limit curve is exceeded.

Proposed Answer: A

Explanation:

A) Correct. The prevention of damage to the tail pipe discharge lines and the resultant stress on containment is given as the reason for this limit in the training document for LGA-003, Page 35.

B) Incorrect. Containment pressure instruments are well above the SRV Tail Pipe Level limit and thus are not a reason for the SRV Tail Pipe Level limit. Plausible as damage to containment and instrument lines is specifically mentioned in the training documents for LGA-003.

C) Incorrect. A containment flood level of 722 ft is the elevation of the lowest suppression chamber-to-drywell vacuum breaker opening. If the vacuum breaker openings are submerged, the vacuum breakers cannot function as designed to relieve non-condensables into the drywell and equalize drywell and suppression chamber pressures. 722 ft is slightly above the SRV Tail Pipe Level Limit Curve (Detail S of LGA-003). Plausible if the candidate confuses the Tail Pipe Curve with the need to maintain the vacuum breakers above the water level.

D) Incorrect. If the Pressure Suppression Pressure limit curve is exceeded it does require an emergency depressurization per LGA-004 and it is a function of Suppression Pool Water Level. [Page 16 of the training for LGA-003]. Plausible as the Pressure Suppression Pressure limit curve is included on LGA-003 (Detail P) and could be confused by the candidate with the Detail S SRV Tail Pipe limit curve.

Technical Reference(s): LGA-003, Primary Containment Control, Rev 18; LGA-003, Primary Containment Control, Training Manual, (p. 34-35) Rev 25

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 422.00.01, Given plant conditions and LGA entry, recall the basis for each portion of the SRV Tail Pipe Limit curve and identify actions when limit is exceeded, while operating the plant or on an exam, IAW the LGA procedures.

Question Source:	Bank #	X	
	Modified Bank #	_	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	River Bend 2014 NRC ILE (Q26)	
Question Cognitive Level:	Memory or Fundamental Knowledge	2	
	Comprehension or Analysis	_	
10 CFR Part 55 Content:	41.5 / 45.6		

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295035 EA1.02	
	Importance Rating	3.8	

K/A Statement: Ability to operate and/or monitor the following as they apply to secondary containment high differential pressure: SBGT/FRVS

Proposed Question: **Question #27**

Unit 2 was in Mode 2 following a refueling outage when the following occurred:

- 2PM06J-A408, RB EXH FANS NO AIR FLOW, is LIT
- 2PM06J-A305, RB 2A VENT EXH FAN AUTO TRIP, Is LIT
- 2PM06J-A407, RB 2C VENT EXH FAN AUTO TRIP, Is LIT
- 2VR04YA/B and 2VR05YA/B are OPEN
- Reactor building vacuum indicates -0.1 inches WC and degrading
- The crew has entered LOA-VR-201, “Unit 2 Recovery from a Group IV Isolation or Spurious Trip of Reactor Building Ventilation”

Based on the above conditions, SBGT ___(1)___ in order to ___(2)___.

A. (1) is manually started
(2) maintain negative pressure in the reactor building

B. (1) is manually started
(2) mitigate airborne contamination

C. (1) automatically started
(2) maintain negative pressure in the reactor building

D. (1) automatically started
(2) mitigate airborne contamination

Proposed Answer: A

Explanation:

One of the purposes of SBGT is to maintain negative pressure in the reactor building. With a spurious trip of reactor building ventilation, the pressure in the reactor building will rise. When the crew implements LOA-VR-201 and with reactor building pressure at 0” WC, the crew will manually start SBGT.

A) Correct. SBGT does not automatically start on high reactor building d/p. Instead the crew must manually start SBGT in order to maintain negative pressure in the reactor building.
B) Incorrect. Plausible as the first part of the distractor is correct. The second part is plausible because a purpose of SBGT is to remove airborne contamination during a fuel handling accident or LOCA. Based on the given conditions, there is no indication of a LOCA or fuel handling accident, so SBGT is started to maintain negative pressure.
C) Incorrect. Plausible as SBGT will automatically start when fuel pool vent radiation monitor is at 10 mrem/hr on Unit 2, or reactor building ventilation radiation monitor reaches 8 mrem/hr. However, there is no automatic initiation of SBGT for high reactor building pressure. The second part is correct, as the reason for it being manually started is to maintain negative pressure in the reactor building.
D) Incorrect. Plausible as SBGT will automatically start when fuel pool vent radiation monitor is at 10 mrem/hr on Unit 2, or reactor building ventilation radiation monitor reaches 8 mrem/hr. However, there is no automatic initiation of SBGT for high reactor building pressure. The second part is plausible because a purpose of SBGT is to remove airborne contamination during a fuel handling accident or LOCA. Based on the given conditions, there is no indication of a LOCA or fuel handling accident, so SBGT is started to maintain negative pressure.

Technical Reference(s): VG-1; LOA-VR-201, Rev 13; LGA-002, Rev 010; 095 Standby Gas Treatment; 118 Rx Bldg HVAC

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 095.00.05, 118.00.05a

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

Question History: Last NRC Exam NA

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis 2

10 CFR Part 55 Content:
55.41 7
55.43 ______

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000 G2.2.39	
	Importance Rating	3.9	

K/A Statement: Knowledge of less than or equal to one hour Technical Specification action statements for systems. (RHR, LPCI) Injection Mode

Proposed Question: **Question 28**

Unit 1 is in Mode 1:

- 1B RHR Pump is OOS
- HPCS is INOP

What is the MAXIMUM time before LCO 3.0.3 is required to be entered per Technical Specification 3.5.1 ECCS- Operating?

A. IMMEDIATELY

B. 20 minutes

C. 30 minutes

D. 1 hour

Proposed Answer: A

Explanation:

A) Correct. TS LCO 3.5.1 Condition H for an inoperable HPCS and one or more low pressure ECCS injection/spray subsystems inoperable requires entry in LCO 3.0.3 IMMEDIATELY. In addition, the RCIC system must be verified by administrative means immediately as well per Condition B.

B) Incorrect. Plausible because TS 3.1.5 has a 20 minute required action time if 2 or more control rod accumulators are inoperable with reactor pressure ≥ 900 psig and charging water pressure < 940 psig.

C) Incorrect. Plausible because 30 minutes is a completion time associated with LCO 3.4.11.

D) Incorrect. Plausible because a completion time of 1 hour is listed in TS 3.0.3 to initiate action towards shutting down the unit. Due to the conditions presented, LCO 3.0.3 must be entered immediately, then have 1 hour to initiate action.

Technical Reference(s):

- Technical Specifications 3.1.5, 3.4.11, 3.5.1 and 3.0.3;
- 064, Residual Heat Removal, Rev 17

Proposed references to be provided to applicants during examination: Tech Spec 3.5.1 with above the line information and less than or equal to 1 hour required actions and completion times blank

Learning Objective: 064.00.022: Given a copy of Technical Specifications key system parameters, and various plant conditions, determine if Technical Specifications LCOs are met, recall the basis for the LCO, and identify the required actions in accordance with Technical Specifications, while operating the system or on an exam.

Question Source:	Bank #	___	
	Modified Bank #	___	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	2	
	Comprehension or Analysis	___	
10 CFR Part 55 Content:	41.7 / 41.10 / 43.2 / 45.13		

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	205000 K1.05	
	Importance Rating	3.1	

K/A Statement: Knowledge of the physical connections and/or cause-effect relationships between shutdown cooling system (RHR shutdown cooling mode) and the following: Component cooling water systems.

Proposed Question: **Question #29**

Unit 1 is in Mode 4 following a refueling outage. The 1B loop of RHR has just been placed in Shutdown Cooling mode with the following conditions:

- Reactor water level is +50 inches and STABLE
- 1B RHR Pump flow is 4000 gpm
- 1A RR loop flow is 3400 gpm
- RHR Service Water flow to the 1B RHR Heat Exchanger is 9500 gpm

What is the current status of the 1B RHR Shutdown Cooling system?

A. 1B RHR Pump flow is causing thermal stratification to occur

B. 1A RR Pump flow indicates the RR pump is dead headed

C. 1B RHR Heat Exchanger service water flow exceeds design limits

D. 1B RHR Flow is NOT high enough to ensure the Min Flow Valve stays CLOSED

Proposed Answer: C

Explanation:

A) Incorrect. Plausible because thermal stratification is possible in this condition, but with RPV level at +50 inches and an RR pump running in the 'A' loop this condition is prevented. Precaution C.18.2 in LOP-RH-07 lists the requirements for adequate core flow to prevent thermal stratification.
B) Incorrect. Plausible as limitation D.5 in LOP-RH-07 states the RHR pump may deadhead by an RR pump. To prevent this, minimum permissible RR pump flow is 3000 gpm and maximum permissible RHR pump flow is 6000 gpm.
C) Correct. Limitation D.5 in LOP-RH-05 and the preceding note both state the maximum service water design flow for the RHR Heat Exchangers is less than 9250 gpm. The conditions state the flow is 9500 gpm, therefore the service water flow exceeds the design limits of the RHR Heat Exchanger.
D) Incorrect. Plausible as an RHR pump with a flow <2000 gpm will cause its minimum flow valve to come open and will result in RCS inventory being sent directly to the suppression pool. In this case RHR flow is the lowest permissible value for the shutdown cooling procedure at 4000 gpm, but is high enough to keep the min flow valve closed.

Technical Reference(s): LOP-RH-05 Rev 34, LOP-RH-07 Rev 80

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 064.00.16a, 064.00.20, 064.05.20

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

Question History: Last NRC Exam LaSalle 2012 NRC ILT EXAM (Q14)

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis 3

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

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 K2.01	
	Importance Rating	3.0	

K/A Statement: Knowledge of electrical power supplies to the following: Pump power

Proposed Question: **Question 30**

The 135X/Y Feed Breaker, 1AP04E-9, at Bus 141Y failed open. What component is affected by this condition?

A. 1B Standby Liquid Control Pump, 1C41-C001B

B. Reactor Recirculation HCU, Sub Loop 1B Motor, 1B33-D003B

C. Reactor Building Equipment Drain Pump, 1RE01P

D. LPCS Water Leg Pump, 1E21-C002

Proposed Answer: D

Explanation:

A) Incorrect. Plausible as the 1B SBLC pump is powered by 480 VAC supply 136Y-2.

B) Incorrect. Plausible as the Reactor Recirculation HCU, Sub Loop 1B Motor is powered by 480 VAC supply 136Y-2.

C) Incorrect. Plausible as the Reactor Building Equipment Drain Pump is powered by 480 VAC supply 133-2.

D) Correct. Powered by 480 VAC power supply 135Y-2.

Technical Reference(s):

- LOP-AP-141Y, "Preparation Procedure for De-Energizing U-1 DIV 1 Busses," Rev 24;
- LOP-AP-142Y, "Preparation Procedure for De-Energizing U-1 DIV 2 Busses," Rev 15;
- Lesson Plan for System 63, Rev 13, Low Pressure Core Spray;
- Training Drawing AP-1, Rev 3, AC Distribution
- LOP-AP01E, "Unit 1 Auxiliary Power System Electrical Checklist," Rev 8

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 063.00.05: Recall the function, theory of operation, interlocks, trips, and characteristics of the following Low Pressure Core Spray System components and relate these items to overall system operation while operating the system or on an exam in accordance with the student text: b. Water Leg Pump.

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

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge 4
Comprehension or Analysis ___

10 CFR Part 55 Content: 41.2 to 41.9 / 45.7 to 45.8

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 K6.02	
	Importance Rating	3.8	

K/A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the low pressure core spray system: Emergency generators

Proposed Question: **Question #31**

Unit 1 is at 100% RTP with the ‘0’ EDG OOS.

At 0800, the following plant parameters were observed on Unit 1 due to a LOCA:

- RPV pressure is 400 psig and LOWERING
- Reactor water level is -150 inches and LOWERING
- Drywell pressure is 2.0 psig and RISING

At 0815, a LOOP occurred on Unit 1.

Which of the following describes the status of the LPCS injection valve at 0820?

A. CLOSED with no flow

B. CLOSED with flow through minimum flow valve

C. OPEN with no flow

D. OPEN with flow through injection valve

Proposed Answer: C

Explanation:

The ‘0’ EDG supplies emergency power to the LPCS system, including the pumps and valves. The given conditions indicate LPCS has received a valid initiation signal, and the LPCS pump will start. Based on pressure, the LPCS Injection Valve will open. With a LOOP and the ‘0’ EDG OOS, the LPCS pump will trip, but the injection valve will remain open, as MOVs fail as is.

A) Incorrect. Plausible if the applicant is not aware of the LPCS initiation setpoints, or does not recognize the setpoints have been reached. Also plausible if the applicant believes the LPCS Injection Valve isolates on a loss of power.
B) Incorrect. Plausible because when LPCS gets an initiation signal based on vessel level and drywell pressure, the LPCS pump will start but the injection valve will not open until RPV pressure is low enough, and the LPCS pump will pump through the minimum flow valve. In these conditions, RPV is low enough for the LPCS Injection valve to open. Also, with a LOOP and the ‘0’ EDG OOS, the LPCS pump will trip and the ‘0’ EDG will not re-energize the bus.
C) Correct. Based on the given conditions, there is a valid LPCS initiation signal, and RPV pressure is low enough that the LPCS Injection Valve will open. When the LOOP occurs and with the ‘0’ EDG OOS, the injection valve will remain open, and the LPCS pump will trip and there will be no LPCS flow.
D) Incorrect. Plausible as the LPCS Injection Valve will open. If the ‘0’ EDG was operable, then the LPCS pump would restart when the ‘0’ EDG re-energizes the bus.

Technical Reference(s): 063 Low Pressure Core Spray, LP-1

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 063.00.02, 063.00.08, 063.05.08

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

Question History: Last NRC Exam NA

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis 2

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

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209002 K3.01	
	Importance Rating	3.9	

K/A Statement: Knowledge of the effect that a loss or malfunction of the high pressure core spray system will have on the following: Reactor water level.

Proposed Question: **Question 32**

HPCS spuriously initiated on a malfunction of the High Drywell Pressure Instrumentation.

- The HPCS Injection Valve (1E22-F004) was MANUALLY OVERRIDDEN by taking the control switch to CLOSE prior to level 8 being reached

Subsequently,

- Reactor level reached level 8 (+55 inches) and then began lowering
- Current reactor level is +35 inches and slowly lowering

Which one of the following is correct concerning the operation of the HPCS Injection Valve (1E22-F004)?

A. The valve can ONLY be opened if the High-Level seal-in is manually RESET and the valve hand switch is placed in the OPEN position on 1H13-P601

B. The valve can be opened by taking the valve hand switch to the OPEN position ONLY

C. The valve will automatically open if the HPCS Manual Initiation Pushbutton on 1H13-P601 is depressed

D. The valve will automatically open if RPV water level drops to Level 2 (-48") as indicated on wide range level indication

Proposed Answer: ___A___

Explanation:

A) Correct. The valve can be manually opened any time a reactor high water level signal is reset. If an initiation signal is present and the automatic function of the valve is overridden, the valve is prevented from automatically reopening until the initiation logic is reset.

B) Incorrect. Plausible if the applicant believes that the valve being overridden negates the high-level seal-in logic.

C) Incorrect. Plausible if the applicant believes that the manual initiation logic negates the high-level seal-in logic.

D) Incorrect. Plausible as this would be the case if the 1E22-F004 was not overridden closed.

Technical Reference(s): 065 HPCS, (p. 9, 25) Rev 9; 1E-1-4222AB/AC/AD

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 061.00.05a; Recall the function, theory of operation, interlocks, trips, and characteristics of the following HPCS system components and relate these items to overall system operation while operating the system or on an exam in accordance with the student text: a. Injection Valve

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

Question History: Last NRC Exam River Bend 2000 NRC ILE (Q51)

Question Cognitive Level:	Memory or Fundamental Knowledge	____
	Comprehension or Analysis	_3_

10 CFR Part 55 Content: 41.7 / 45.4

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209002 A1.04	
	Importance Rating	3.3	

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operating the high pressure core spray system controls including: Reactor pressure

Proposed Question: **Question #33**

Unit 1 was operating at 100% RTP with the MDRFP OOS when the following occurred:

- 1PM03J-A106, 1A/1B/1C FW PMP SUCT PRESS LO is LIT
- The reactor was manually scrammed
- 1H13-P601-A208, RX VESSEL WTR LVL 2 LO-LO is LIT
- 1H13-P601-A308, RX VESSEL WTR LVL 2 LO-LO is LIT
- Reactor pressure is 915 psig

IMMEDIATEY after the crew observes the above conditions, the HPCS Pump is ______.

A. running and injecting at approximately 2000 gpm

B. running and injecting at approximately 6250 gpm

C. NOT running and must be manually started

D. NOT running and should be left in standby

Proposed Answer: A

Explanation:

The HPCS system automatically initiates on low vessel level (Level 2) and/or high drywell pressure. With the loss of feedwater and reactor vessel level lowering to Level 2, HPCS will automatically initiate. Under these conditions, the RPV is intact, and reactor pressure will not lower and drywell pressure will not increase. From lesson plan 061 High Pressure Core Spray, HPCS flow at the given reactor pressure is approximately 2000 gpm.

A) Correct. Reactor water level has reached Level 2, and the HPCS system will receive an initiation signal which will start the HPCS pump. Since the RPV is at 915 psig, it will inject at approximately 2000 gpm.
B) Incorrect. Plausible as 6250 gpm is the rated flow for the HPCS Pump. However, it will not reach rated flow until the d/p between the RPV and HPCS pump suction is 370 psid. Under these conditions, there has only been a loss of feedwater, and the RPV is not being depressurized.
C) Incorrect. Plausible because the HPCS Pump can be manually started from the control room. If the auto initiation of HPCS failed, then HPCS would need to be manually started.
D) Incorrect. The HPCS Pump is normally in standby and ready to receive an initiation signal. Therefore, this is plausible if the examinee believes the auto initiation setpoints have not been reached for vessel level or believes HPCS only initiates on high drywell pressure and believes that HPCS injection is not needed for this condition.

Technical Reference(s): 061 High Pressure Core Spray pages 7, 15, and 16; HP-1; LOR-1H13-P601-A208

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 061.00.06f, 061.00.07a, 061.00.07b

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

Question History: Last NRC Exam NA

Question Cognitive Level:
- Memory or Fundamental Knowledge ____
- Comprehension or Analysis 2

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

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000 K4.07	
	Importance Rating	3.8	

K/A Statement: Knowledge of standby liquid control system design feature(s) and/or interlocks which provide for the following: RWCU isolation

Proposed Question: **Question 34**

Unit 1 is in an ATWS condition:

- The SBLC Pump keylock switch is placed in 'SYS A' at 1H13-P603.

This will cause the ‘1A’ SBLC Pump to start AND…

NOTE: RWCU Inboard Isolation Valve 1G33-F001
RWCU Outboard Isolation Valve 1G33-F004

A. BOTH System ‘A’ AND System ‘B’ Squib valves will energize AND BOTH 1G33-F001 AND 1G33-F004 will CLOSE

B. ONLY the System ‘A’ squib valve will energize AND 1G33-F004 will CLOSE

C. BOTH System ‘A’ AND System ‘B’ Squib valves will energize AND 1G33-F004 will CLOSE

D. ONLY the System ‘A’ squib valve will energize AND BOTH 1G33-F001 AND 1G33-F004 will CLOSE

Proposed Answer: C

Explanation:

A) Incorrect. Plausible as both trains Squib valves will energize. The RWCU isolation is plausible as this would be the case for a Group 5 containment isolation. Only the outboard valve closes for an SBLC initiation.

B) Incorrect. Plausible as the applicant may believe that only the train A Squib valve will energize when only the A SBLC pump is started. The RWCU isolation is correct.

C) Correct. When the SBLC key lock switch is placed in either SYS A or SYS B both trains Squib valves will fire and the RWCU Outboard isolation valve 1G33-F004 will CLOSE to prevent the boron added from being removed by the RWCU Filter/Demineralizer components.

D) Incorrect. Plausible as the applicant may believe that only the train A Squib valve will energize when only the A SBLC pump is started. The RWCU isolation is plausible as this would be the case for a Group 5 containment isolation. Only the outboard valve closes for an SBLC initiation.

Technical Reference(s): 028, SBLC, (p. 23) Rev 12

Proposed references to be provided to applicants during examination: N/A

Learning Objective: Objective #028.00.05: Recall the function, theory of operation, interlocks, trips, and characteristics of the following Standby Liquid Control System components and relate these items to overall system operation while operating the system or on an exam in accordance with the student text: n. Key Lock Initiation switches

Question Source:	Bank #	___	
	Modified Bank #	X	(Note changes or attach parent)
	New	___	
Question History:	Last NRC Exam	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	___	
	Comprehension or Analysis	2	
10 CFR Part 55 Content:	41.7		

Comments: Modified from LaSalle exam bank question **753137**.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000 A2.03	
	Importance Rating	3.2	

K/A Statement: Ability to (a) predict the impacts of the following on the standby liquid control system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: AC power failures

Proposed Question: **Question #35**

Unit 2 is operating at 100% RTP with the SBLC in standby when the following occurred:

- 2H13-P603-B502, STANDBY LIQ TANK TEMP HI/LO, is LIT
- 2PM01J-A412, 480V SWGR 235X/Y MAN OPER BKR AUTO TRIP, is LIT
- SBLC Tank Temperature at 2H22-P011 indicates 68°F
- MCC 235Y-1 Breaker C1, “A SBLC Storage Tank Heater” is tripped and CANNOT be reset

In accordance with the alarm response manual LOR-2H13-P603-B502, the ‘B’ SBLC Storage Tank heater ____(1)____ in order to prevent ____(2)____ in the event SBLC injection is required.

A. (1) automatically started
(2) a positive reactivity addition from cold SBLC water

B. (1) automatically started
(2) boron precipitation

C. (1) must be manually started locally
(2) a positive reactivity addition from cold SBLC water

D. (1) must be manually started locally
(2) boron precipitation

Proposed Answer: D

Explanation:

The SBLC Storage Tank contains two heaters to maintain the tank at a temperature that won't result in boron precipitation (basis for TS 3.1.7). The alarm setpoint of low temperature (70 deg. F) will annunciate to prompt operators to take action to restore heating to the tank. The 'A' heater automatically cycles on and off from 75 deg. F to 85 deg. F, and the 'B' heater will only energize when the local handswitch is taken to the 'B ON' or 'A/B ON' position (028 SBLC). From the given information, the SBLC system is in standby, meaning the heater handswitch is in the 'AUTO' position. With the AC power failure to the 'A' heater, the tank temperature must be restored by manually starting the 'B' heater locally.

A) Incorrect. Plausible because the 'A' heater would have automatically energized on low tank temperature, but incorrect because the 'B' heater must be started locally. The second part is plausible because cold water is a positive reactivity addition, but with the high concentration of boron in the SBLC tank, the SBLC tank temperature has a negligible impact on reactivity.
B) Incorrect. Plausible because the 'A' heater would have automatically energized on low tank temperature, but incorrect because the 'B' heater must be started locally. The second part of the distractor is correct, boron may precipitate out of solution at lower tank temperatures.
C) Incorrect. Plausible because the first part of the distractor is correct, the 'B' heater must be manually started locally. The second part is incorrect but plausible because cold water is a positive reactivity addition, but with the high concentration of boron in the SBLC, the SBLC tank temperature has a negligible impact on reactivity.
D) Correct. The 'B' heater must be manually started locally, and boron may precipitate out of solution at lower tank temperatures.

Technical Reference(s): 2H13-P603-B502 Rev. 003; 028 Standby Liquid Control, pages 6, 15; LOP-AP-241Y Rev. 014, page 32, B 3.1.7 Standby Liquid Control System Rev. 070, page 4; SC-1

Proposed references to be provided to applicants during examination:

Learning Objective: 028.00.06o, 028.00.15, 028.00.16

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

Question History: Last NRC Exam NA

Question Cognitive Level:
Memory or Fundamental Knowledge 3
Comprehension or Analysis ____

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

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 K5.02	
	Importance Rating	3.3	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to the reactor protection system: Specific logic arrangements

Proposed Question: **Question 36**

Concerning the Reactor Protection System (RPS) scram logic, a trip of ____(1)____ and ___(2)___ subchannels will cause a ____(3)_____ reactor scram.

A. (1) A1
(2) A2
(3) FULL

B. (1) A1
(2) B1
(3) HALF

C. (1) A1
(2) B1
(3) FULL

D. (1) A2
(2) B1
(3) HALF

Proposed Answer: C

Explanation:

A) Incorrect. Plausible as a trip of this logic arrangement will result in RPS generating a half SCRAM on RPS A.

B) Incorrect. Plausible as a trip of this logic arrangement will result in RPS generating a FULL SCRAM.

C) Correct. RPS logic is one-out-of-two-twice, which means that either A1 OR A2, AND B1 OR B2 will actuate a full reactor scram.

D) Incorrect. Plausible as a trip of this logic arrangement will result in RPS generating a FULL SCRAM.

Technical Reference(s): 049, Reactor Protection System, Rev 11; 1E-1-4215AH/AJ

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 049.00.010: Recall the signals which cause a Reactor Protection System trip, include setpoints, logic, how and when bypassed, and how they are reset while operating the system, or on an exam in accordance with the student text and station procedures

Question Source:	Bank #	X
	Modified Bank #	_
	New	
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge	2
	Comprehension or Analysis	-
10 CFR Part 55 Content:	41.5 / 45.3	

Comments: LaSalle Question Bank **#754484**

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 A3.05	
	Importance Rating	3.9	

K/A Statement: Ability to monitor automatic operations of the reactor protection system including: SCRAM instrument volume level

Proposed Question: **Question #37**

Which of the following would cause an automatic scram with the Reactor Mode Switch placed* in SHUTDOWN?
Removed the word "placed" as was considered confusing to applicants.

A. Spurious MSIV closure

B. Low Charging Water Header Pressure with the Test Switch in NORMAL

C. Short reactor period due to moving the ‘A’ SRM detector

D. High Scram Discharge Volume Level with SDV HI LVL SCRAM BYPASS Switch in NORMAL

Proposed Answer: __D__

Explanation:

During this surveillance with the mode switch in Shutdown, the SDV High Level trip must be bypassed prior to resetting the scram. If it is not bypassed, the reactor will automatically scram on high SDV level when the Scram Reset Switch is taken to GROUP 1/4 then GROUP 2/3.

A) Incorrect. Plausible because spurious MSIV closures will cause RPS to automatically scram the reactor with the mode switch in Run. However, with the mode switch in Shutdown, this automatic scram is bypassed.
B) Incorrect. Plausible as this is a scram signal, but which is bypassed when the Reactor Mode Switch is in SHUTDOWN.
C) Incorrect. Plausible because moving the ‘A’ SRM Detector may cause a short period. However, a short period is only an alarm and there is no automatic scram associated with a short period on the SRM.
D) Correct. The SDV High Level trip remains active with the Reactor Mode Switch in SHUTDOWN unless the SDV High Level Bypass Switch is placed in BYPASS.

Technical Reference(s): LOS-RP-R1, Rev. 021 Attachment 1A; 049 RPS, pages 14, 17, 19, and 27; LOR-1H13-P603-B406, Rev. 001; RD-1

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 049.00.05l

Question Source: Bank # __X__
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam Fermi 2015 ILT Exam (Q33)

Question Cognitive Level: Memory or Fundamental Knowledge __3__
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 __7__
55.43 ____

Comments: Fermi Exam (ML15287A053)

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003 K6.05	
	Importance Rating	3.1	

K/A Statement: For Safety Function 7 (IRM): Knowledge of the affect that a loss or malfunction of the following will have on the intermediate range monitor system: Trip Units

Proposed Question: **Question 38**

Unit 1 is at 2% RTP

- The 1B Intermediate Range Monitor (IRM) is placed in BYPASS to permit troubleshooting
- The maintenance technician incorrectly places the 1E IRM drawer Mode Switch in the STANDBY position

Subsequently:

- The 1A IRM drawer Output Amplifier fails resulting in the loss of input to all 1A IRM trip units

1A and 1E IRMs are BOTH…

A. reading downscale AND a reactor scram has NOT occurred

B. inoperable AND a reactor scram has occurred

C. reading downscale AND a reactor scram has occurred

D. inoperable AND a reactor scram has NOT occurred

Proposed Answer: ___D___

Explanation:

A) Incorrect. The white lights lit on the IRM drawers on 1H13-P635/636 are due to both IRMs 1A & 1E being inoperable (1A due to the loss of the Trip Unit input signals from the Output Amplifier; 1E from the technician placing the Mode Switch in "Standby." Only IRM 1A will have a white "downscale" light lit due to loss of the trip unit.) and not due to both reading downscale. The second part is correct since both INOP IRMs input to RPS Channel A, and a ½ scram of Channel A will result; the reactor will remain online. Plausible because the white lights (L4) will be lit for the given conditions but indicate that the channels are inoperable; it will only take a little misunderstanding to pick this answer.

B) Incorrect. Although the white lights lit on the IRM drawers on 1H13-P635/636 are due to both the IRMs 1A & 1E being inoperable, as noted above, both INOP IRMs input to RPS Channel A, and only a ½ scram of Channel A will result; the reactor will remain online. Plausible if the candidate confuses which IRMs input to which RPS Channels; it will only take a little misunderstanding to pick this answer.

C) Incorrect. This distractor is counting on a misunderstanding regarding the white lights on the back panels. Also, as noted above, the reactor will not scram. Plausible if the candidate confuses the meaning of the white lights and which IRMs input to which RPS Channels.

D) Correct. Both IRM 1A & 1E are inputs to RPS Channel A; only a ½ scram of Channel A will result and the reactor will not scram. The white INOP lights will be lit on both IRM 1A & 1E drawers. (1A in response to the loss of the Trip Units input signals from the Output Amplifier; 1E in response to the technician placing the Mode Switch in "Standby.")

Technical Reference(s):

- Licensee Event Report 05000219/2004-003, Oyster Creek
- Lesson Plan for System 42, Rev 7, Intermediate Range Monitors;

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 042.00.021: Given a System lineup and various plant conditions, predict the Intermediate Range Monitor System response to various system component failures while operating the system or on an exam in accordance with student text.

Question Source:	Bank #	___	
	Modified Bank #	___	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	__	
	Comprehension or Analysis	3	

10 CFR Part 55 Content: 41.7 / 45.7

Comments:

Excerpt from Oyster Creek LER

On May 27, 2004, at 00:31 hours, with the Reactor Mode switch in the Startup position, a reactor scram from approximately 2% power was caused by a spurious actuation of Nuclear Instrumentation (NI) Intermediate Range Monitor (IRM) (EIIS-IG) channels 13, 14, and 18. The spurious actuation was caused by electromagnetic interference (EMI). The reactor shut down as designed.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003 A4.03	
	Importance Rating	3.6	

K/A Statement: Ability to manually operate and/or monitor in the control room: IRM range switches.

Proposed Question: **Question #39**

A reactor startup is in progress on Unit 1 with the following indications:

- At 0800 – IRM Channel 'C' reads 20/40 scale on Range 5
- At 0815 – Reactor power has doubled 3 times

At 0815 the crew should expect IRM Channel 'C' to indicate _____ .

A. 60/125 on Range 6

B. 80/125 on Range 6

C. 16/40 on Range 7

D. 35/40 on Range 7

Proposed Answer: ___C__

Explanation:

From Lesson Plan 042 IRMs, the odd numbered ranges are 0-40, and the even numbered ranges are 0-125. When an IRM is on an odd numbered range, the readings overlap the lower third of the scale for the next even numbered range. For example, 20/40 on Range 5 is 20/125 on Range 6. Also from 042, every even numbered range position indicates a decade above the next even numbered position, and same for the odd numbered range positions. For example, 80/125 on Range 6 is 8/125 on Range 8, and 20/40 on Range 5 is 2/40 on Range 7.

A) Incorrect. Plausible because multiplying the initial 20/40 by three instead of doubling it three times is 60/125 on Range 6.
B) Incorrect. Plausible because the first doubling is an increase of 20. So, if 20 is added for the second and third time instead of doubling the number, the indication is 80/125 on Range 6.
C) Correct. The first doubling is 40/40 on Range 5, which is 40/125 on Range 6. The second doubling is 80/125 on Range 6. The third doubling would be off the scale for Range 6 (160/125), so the IRM is switched to Range 7. From the discussion in the previous paragraph, Range 7 indicates one decade below what the reading would have been on Range 5, which is 16/40 on Range 7, or 16/125 on Range 8.
D) Incorrect. Plausible because the third doubling would read 160/125 on Range 6, which is exceeded by 35 on Range 6. If the applicant assumes the excess is directly translated onto the next range, then Range 7 would read 35/40.

Technical Reference(s): 042 Intermediate Range Monitoring System, pages 5-6

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 042.00.05c

Question Source: Bank # __X
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam Columbia 2006 ILT Exam Q28

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis __3_

10 CFR Part 55 Content: 55.41 _1,7_
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215004 A1.03	
	Importance Rating	3.4	

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operating the source range monitor system controls including: RPS status.

Proposed Question: **Question 40**

Select the following condition(s) associated with the Source Range Monitors (SRM) that will cause a reactor scram signal if the shorting links are REMOVED?

1) SRM High-High trip (5 x 10^5 CPS)
2) SRM Short Period trip (50 seconds)
3) SRM Downscale

A. 1 and 3

B. 2 and 3

C. 2 Only

D. 1 Only

Proposed Answer: D

Explanation:

A) Incorrect. Only a high-high trip will cause an RPS actuation with the shorting links removed. Plausible because an association can be made with a downscale condition there is no indication of reactor power and therefore a SCRAM is needed.

B) Incorrect. Only a high-high trip will cause an RPS actuation with the shorting links removed. Plausible because an association can be made with the importance of a short period condition and the need to terminate such condition with an RPS scram.

C) Incorrect. Only a high-high trip will cause an RPS actuation with the shorting links removed. Plausible because an association can be made with the importance of a short period condition and the need to terminate such condition with an RPS scram.

D) Correct. Only a high-high trip will cause an RPS actuation with the shorting links removed.

Technical Reference(s):

- LOP-NR-01, Rev 16, Source Range Monito Operations;
- Lesson Plan 041, Rev 7, Source Range Monitors:

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 041.00.018: Given various plant conditions, predict the response of the following supported systems to a loss of the Source Range Monitor System while operating the system, or on an exam in accordance with student text: a. Reactor Protection System

Question Source:	Bank #	X	
	Modified Bank #	_	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	2	
	Comprehension or Analysis		
10 CFR Part 55 Content:	41.5 / 45.5		

Comments: LaSalle Question Bank **#1687546**

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005 A2.02	
	Importance Rating	3.6	

K/A Statement: Ability to (a) predict the impacts of the following on the average power range monitor/local power range monitor system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Upscale or downscale trips

Proposed Question: **Question #41**

Unit 1 is performing a shutdown IAW LGP-2-1, "Normal Unit Shutdown."

At 95% RTP, the following occurred:

- 1H13-P603-A407, LPRM DOWNSCALE, is LIT
- On panel 1H13-P608, the downscale light for center core LPRM 32-33D ('A' APRM) is WHITE
- The crew has implemented LOA-NR-101, "Neutron Monitoring Trouble"

Which of the following will the crew perform IAW LOA-NR-101?

1. Stop the unit shutdown
2. Bypass the LPRM
3. Reset the LPRM
4. Bypass the affected APRM
5. Bypass the affected OPRM

A. 3 ONLY

B. 1 and 3 ONLY

C. 1, 2 and 4 ONLY

D. 2, 4, and 5 ONLY

Proposed Answer: __C__

Explanation:

During a reactor shutdown, LPRM downscale lights are expected, but they are not expected to fail downscale. Therefore, the crew has implemented LOA-NR-101. The actions in the procedure are to stop power changes, then bypass the failed LPRM, then bypass the affected APRM. The OPRM is not required to be bypassed unless it becomes inoperable.

A) Incorrect. Plausible because downscale lights are expected to occur during a unit shutdown, but not fail downscale and not the center LPRM at this power level. The first step IAW LOA-NR-101 is to stop all power changes. The white downscale light on a single LPRM indicates the LPRM is failed. Resetting the LPRM will not clear the alarm condition.
B) Incorrect. Plausible because the first step IAW LOA-NR-101 is to stop all power changes. However, a downscale light on a single LPRM indicates the LPRM is failed. Resetting the LPRM will not clear the alarm condition.
C) Correct. IAW LOA-NR-101, the first action is to stop the power change, then to bypass the failed LPRM and bypass the affected APRM.
D) Incorrect. Plausible because actions II and IV are correct, however LOA-NR-101 directs the crew to stop the power change, and the OPRM is not required to be bypassed unless it becomes inoperable.

Technical Reference(s): 043 LPRM pages 14, 16-17, 19; LGP-2-1 Rev. 119, pages 19-23; LOA-NR-101 Rev. 020 pages 6-7; LOR-1H13-P603-A407 Rev. 03

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 043.00.07b

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

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge _3_
Comprehension or Analysis ___

10 CFR Part 55 Content: 55.41 __**5**_
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000 A3.04	
	Importance Rating	3.6	

K/A Statement: Ability to monitor automatic operations of the reactor core isolation cooling system including: System flow

Proposed Question: **Question 42**

Unit 2 has Scrammed from 100% RTP:

- RCIC flow is in AUTOMATIC, injecting at rated flow
- SRV's are being cycled to maintain reactor pressure

Compared to current conditions, what will the FINAL RCIC system parameters be AFTER reactor pressure rises from 800 to 1000 psig?

	Turbine Speed	Pump Flow	Pump Discharge Pressure
A.	Lower	Remain the Same	Higher
B.	Remain the Same	Lower	Lower
C.	Higher	Remain the Same	Higher
D.	Higher	Higher	Remain the Same

Proposed Answer: ___C___

Explanation:

A) Incorrect. Plausible as the applicant could believe that the RCIC turbine speed would lower as it works harder to overcome the higher discharge pressure. The other parameters are correct.

B) Incorrect. Plausible as this would be the case if the pump flow controller was not in automatic maintaining constant flow with RPV pressure increasing.

C) Correct. As RPV pressure rises, pump discharge pressure rises, therefore RCIC pump turbine must draw more steam and thus speed up to overcome the greater discharge pressure to maintain constant flow at the automatic flow controller setpoint.

D) Incorrect. Plausible as this would be the case if pump discharge pressure didn't increase with RPV pressure.

Technical Reference(s): 032, RCIC (p. 58-59) Rev 9

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 032.00.14; Given a RCIC System lineup and various plant conditions, evaluate the following system indications/responses. Determine if the indications/responses are expected and normal while operating the system, or on an exam in accordance with the student text and station procedures: b. Flow

Question Source:	Bank #	_X_
	Modified Bank #	___ (Note changes or attach parent)
	New	____
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory or Fundamental Knowledge	___
	Comprehension or Analysis	__2_
10 CFR Part 55 Content:	41.7, 45.7	

Comments: LaSalle Question Bank **#784713**

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	218000 A4.01	
	Importance Rating	4.4	

K/A Statement: Ability to manually operate and/or monitor in the control room: ADS valves.

Proposed Question: **Question #43**

A LOCA occurred on Unit 2 and the following conditions exist:

- Reactor Pressure is 475 psig and slowly LOWERING
- Reactor Water Level is -150 inches and STABLE
- Drywell Pressure is 0.5 psig and STABLE (Max pressure reached was 2.0 psig)
- ADS automatically initiated and all ADS valves are OPEN

How will the ADS valves respond if BOTH divisional LOW RX LVL RESET pushbuttons are simultaneously depressed and released?

A. Remain OPEN

B. CLOSE and stay CLOSED

C. CLOSE, then OPEN immediately

D. CLOSE, then OPEN in approximately 118 seconds

Proposed Answer: D

Explanation:

The Low Level Reset Pushbuttons will reset both 118 second timers, therefore the valves will go close. However, the valves will not stay closed because the high drywell pressure signal must also be reset, therefore the ADS valves will re-open in about 118 seconds.

A) Incorrect. Plausible because the low level condition has not cleared, and it is plausible to believe the ADS valves will remain open until the low level condition is cleared if the low level reset pushbuttons are depressed. However, depressing the low level reset pushbuttons will reset the 118 second timers and the ADS valves will go closed if the low level condition does not need to be corrected.
B) Incorrect. Plausible as the ADS valves will initially go closed and the conditions indicate the high drywell pressure condition is clear. However, to remain closed, the high drywell pressure signal must also be reset.
C) Incorrect. Plausible because the ADS valves will go closed after the Low Level Reset Pushbuttons are depressed and reset the 118 second timers. Since the high drywell pressure reset pushbuttons were not depressed, the ADS valves will re-open after 118 seconds, but not immediately.
D) Correct. The ADS valves will go closed after the Low Level Reset Pushbuttons are depressed and reset the 118 second timers. Since the high drywell pressure reset pushbuttons were not depressed, the ADS valves will re-open after 118 seconds.

Technical Reference(s): LOP-MS-03, Rev 011, pp 3-4; 062 Automatic Depressurization System p 13; NB-1

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 062.00.14

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

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis 3

10 CFR Part 55 Content: 55.41 7
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	223002 G2.1.28	
	Importance Rating	4.1	

K/A Statement: Knowledge of the purpose and function of major system components and controls. Primary Containment Isolation/Nuclear Steam Supply Shutoff

Proposed Question: **Question 44**

The time required to CLOSE the Main Steam Isolation Valves (MSIVs) is SHORT enough to ___(1)___, and with a PCIS signal the MSIV containment isolation circuits are ___(2)___.

A. (1) minimize the loss of coolant from a steam line break
(2) DEENERGIZED

B. (1) prevent exceeding regulatory requirements for offsite release
(2) DEENERGIZED

C. (1) minimize the loss of coolant from a steam line break
(2) ENERGIZED

D. (1) prevent exceeding regulatory requirements for offsite release
(2) ENERGIZED

Proposed Answer: A

Explanation:

A) Correct. The time required to close the Main Steam valves shall be short enough to minimize the loss of coolant from a steam line break. During normal plant operation, the isolation control system sensors and trip controls that are essential to safety are energized. When abnormal conditions are sensed, trip channel sensor contacts open and cause contacts in the trip logic to open and initiate isolations. De-energizing the circuit results in isolation.

B) Incorrect. Plausible as containment isolation is performed typically to reduce to the spread of radioactive contamination during accident scenarios. Part 2 is correct as PCIS logic circuits should not require power to be effective at performing their safety function. PCIS associated with RCIC is the exception to this rule.

C) Incorrect. Plausible as Part 1 is correct and there are PCIS circuits which are required to energize when actuated associated with RCIC and RHR steam condensing valves.

D) Incorrect. Plausible as containment isolation is performed typically to reduce to the spread of radioactive contamination during accident scenarios, and there are PCIS circuits which are required to energize when actuated associated with RCIC and RHR steam condensing valves.

Technical Reference(s): 091, Primary Containment Isolation System (p. 2-3) Rev 15

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 091.00.01; Recall the design bases while operating the Primary Containment Isolation System or on an exam in accordance with the UFSAR and procedures/student text.

Question Source:	Bank #	___	
	Modified Bank #	___	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	_	
	Comprehension or Analysis	2	
10 CFR Part 55 Content:	41.7		

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	239002 K2.01	
	Importance Rating	2.8	

K/A Statement: Knowledge of the electrical power supplies to the following: SRV solenoids.

Proposed Question: **Question #45**

Which of the following DC distribution panels, if de-energized, will prevent the MANUAL operation of ALL individual SRVs at panel 2H13-P601?

A. 211X

B. 211Y

C. 212X

D. 212Y

Proposed Answer: B

Explanation:

The SRVs have two modes of operation – SRV Safety Mode and SRV Relief Mode. In safety mode, the SRVs will lift mechanically based on RPV pressure. In relief mode, the solenoids are energized, admitting air to pneumatically open the SRVs. On a loss of 125 VDC Div 1 or DC Dist. Panel 211Y, the solenoids lose power, so the SRVs cannot be individually opened, though the ADS valves can be opened for pressure control.

A) Incorrect. Plausible because 211X is also powered from 125 VDC Div 1, though there are no components that affect the SRVs from this panel.
B) Correct. On a loss of panel 211Y, the solenoids lose power and the SRVs cannot be individually opened from panel 2H13-P601.
C) Incorrect. Plausible because this panel is powered from 125 VDC Div 2, which powers ADS Div II initiation logic, however this panel, as with 211X, has no components that affect SRVs.
D) Incorrect. Plausible because the loss of 212Y will de-energize ADS Div II initiation logic, however all SRVs can still be manually opened.

Technical Reference(s): 070 Main Steam, pp 8-12, 24; LOA-DC-201 Rev. 021, pp 39, 85; MS-2

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 070.00.05b

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

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge 3
Comprehension or Analysis ___

10 CFR Part 55 Content: 55.41 **2-9**
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	259002 K1.15	
	Importance Rating	3.2	

K/A Statement: Knowledge of the physical connections and/or cause-effect relationships between Reactor Water Level Control System and the following: Recirculation flow control system

Proposed Question: **Question 46**

Unit 1 was operating at 100% RTP:

- RPV water level is +5 inches and LOWERING due to low suction pressure trip on all Feedwater Pumps

Which of the following describes the CURRENT status of the Reactor Recirculation System?

A. Recirculation Pumps are in SLOW speed operation and the Recirculation Flow Control Valves have runback

B. Recirculation Pumps are in SLOW speed operation and the Recirculation Flow Control Valves remain at the pre-transient position

C. Recirculation Pumps are OFF and the Recirculation Flow Control Valves have runback

D. Recirculation Pumps are OFF and the Recirculation Flow Control Valves remain at the pre-transient position

Proposed Answer: A

Explanation:

A) Correct. The EOC-RPT and RPV level less than 12.5 inches results in a downshift to slow speed for the RR pumps. The RR FCVs will runback to 15% with available FW flow < 5% greater than steam flow and level less than Level 4.

B) Incorrect. Plausible as Part 1 is correct and a high DW pressure causes FCVs to remain in their pre-transient position.

C) Incorrect. Plausible as the applicant could believe that the EOC-RPT is a pump trip and not a pump downshift as its name would imply. The RR pumps would downshift on EOC-RPT with reactor power above 25% and a close of the turbine stop valves. RPV level < 12.5" (Level 3) would also result in a downshift of the RR pumps. Part 2 is correct as RR FCVs will runback to 15% with available FW flow < 5% greater than steam flow and level less than Level 4.

D) Incorrect. Plausible as the applicant could believe that the EOC-RPT is a pump trip and not a pump downshift as its name would imply. The RR pumps would downshift on EOC-RPT with reactor power above 25% and a close of the turbine stop valves. RPV level < 12.5" (Level 3) would also result in a downshift of the RR pumps. Plausible as a high DW pressure causes FCVs to remain in their pre-transient position.

Technical Reference(s): 023, Recirculation Flow Control, (p. 32) Rev 8

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 023.05.10; Recall the signals which cause a RRFC System trip, including setpoints and how reset, and predict system response while operating the system on an exam in accordance with the student text and station procedures.

Question Source:	Bank #	X	
	Modified Bank #	___ (Note changes or attach parent)	
	New	___	
Question History:	Last NRC Exam	River Bend 2000 NRC ILE (Q81)	
Question Cognitive Level:	Memory or Fundamental Knowledge		___
	Comprehension or Analysis		3
10 CFR Part 55 Content:	41.2 to 41.9 / 45.7 to 45.8		

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	261000 K3.03	
	Importance Rating	3.2	

K/A Statement: Knowledge of the effect that a loss or malfunction of the standby gas treatment system will have on the following: Primary containment pressure.

Proposed Question: **Question #47**

A small LOCA occurred on Unit 1 and the following conditions exist:

- Unit 1 Standby Gas Treatment (VG) is lined up to vent the Primary Containment IAW LGA-VQ-101, “Unit 1 Containment Vent”
- Containment pressure begins to lower

ONE minute later, containment pressure STOPS LOWERING.

Which of the following could explain the reason containment pressure stopped LOWERING?

A. Trip of the VG Heater

B. RPV level reached level 3

C. Loss of Instrument Air

D. Loss of power to 112Y

Proposed Answer: D

Explanation:

A LOCA has occurred in the drywell, and SBGT automatically started, and the crew has aligned VG to vent containment to lower pressure IAW LGA-VQ-101. In this alignment, when VG is lost, containment pressure will stop lowering.

A) Incorrect. Plausible because the VG heater will trip when there is not adequate VG flow, but the trip of the VG heater by itself will not cause the VG fan to trip.
B) Incorrect. Plausible because several systems isolate on RPV Level 3, but VQ isolates at RPV level 2.
C) Incorrect. Plausible instrument air provides the motive force for VG components, but it would take more than a minute for air pressure to bleed off.
D) Correct. Loss of power to system initiation logic will cause the operating SBGT train to stop operating, therefore causing Primary Containment pressure to rise.

Technical Reference(s): 095 Standby Gas Treatment, pp 6, 14, 15-16, 19; LGA-VQ-101 Rev. 001, pp 14-15; VG-1

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 095.00.21

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

Question History: Last NRC Exam NA

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis 3

10 CFR Part 55 Content:
55.41 7
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001 K4.03	
	Importance Rating	3.1	

K/A Statement: Knowledge of AC electrical distribution design feature(s) and/or interlocks which provide for the following: Interlocks between automatic bus transfer and breakers

Proposed Question: **Question 48**

The 12 kV Distribution System is in a normal lineup.

- 12 kV Circuit Breaker from the 138 kV transformer tripped

What is the lineup of the Auto-Throwover (ATO) Device following this transient?

A. Preferred Feed open, Alternate feed closed, Bus-tie closed

B. Preferred Feed closed, Alternate feed closed, Bus-tie open

C. Preferred Feed closed, Alternate feed open, Bus-tie closed

D. Preferred Feed open, Alternate feed closed, Bus-tie open

Proposed Answer: A

Explanation:

A) Correct. From a normal line-up, if TR-71 lost power, and power is available from TR-52, the ATO will automatically Open the Preferred Feed Disconnect and Close the Alternate Feed Disconnect. From this point, the ATO will NOT return to normal if TR-71 power is restored. HOWEVER, as long as a fault does not exist on Bus 161, if the ATO is in auto, and TR-52 loses power, and power is available from TR-71, the ATO will transfer, (power seek), i.e., the Alternate Feed Disconnect will Open and the Preferred Feed Disconnect will Close, Bus Tie remains closed.

B) Incorrect. Plausible if the applicant believes that the alternate feeder breaker closes causing the preferred source to be attached by the Bus-tie opening.

C) Incorrect. Plausible as this would be the condition if the 12 kV distribution system was initially being powered from the alternate power source.

D) Incorrect. Plausible as the preferred and alternate source feeder breaker positions are correct.

Technical Reference(s): 007, 12 Kv Distribution, (p. 14) Rev 25

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 007.00.05i; Recall the function, theory or operation, interlocks, trips and characteristics of the following 12KV System components and relate these items to overall system operation while operating the system or on an exam in accordance with the student text: i. Auto Throw-over Device

Question Source:	Bank #	X	
	Modified Bank #	_	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	4	
	Comprehension or Analysis	_	
10 CFR Part 55 Content:	41.7		

Comments: LaSalle Question Bank **#753844**

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262002 K6.02	
	Importance Rating	2.8	

K/A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the uninterruptible power supply (AC/DC): DC electrical power.

Proposed Question: **Question #49**

If the Unit 2 Process Computer UPS Inverter output is lost, the alternate power supply is supplied from ___(1)___ and the alternate power supply ___(2)___ to restore power.

A. (1) MCC 135X-2
(2) will automatically align

B. (1) MCC 135X-2
(2) must be manually aligned

C. (1) MCC 235X-2
(2) will automatically align

D. (1) MCC 235X-2
(2) must be manually aligned

Proposed Answer: A

Explanation:

The Unit 1 and Unit 2 Process Computers are each powered from a UPS inverter that is powered from AC and from DC. When the inverter is lost due to a loss of both AC and DC power, the back-up power supply automatically aligns via a static switch.

A) Correct. The back-up power supply for Unit 2 comes from MCC 135X-2 from Unit 1, and when the normal UPS inverter is lost, the back-up will automatically align.
B) Incorrect. Plausible as the back-up power supply is correct, but the back-up power supply does not need to be manually aligned, it will automatically align.
C) Incorrect. Plausible as MCC 235X-2 is the back-up power supply to Unit 1. The second part of the distractor is correct, as the back-up power supply automatically aligns.
D) Incorrect. Plausible as MCC 235X-2 is the back-up power supply to Unit 1, but the back-up power supply does not need to be manually aligned, it will automatically align.

Technical Reference(s): 050 Process Computer, page 42; LOA-DC-201 Rev. 021, pp 111 and 116

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 050.00.05, 050.00.06

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

Question History: Last NRC Exam LaSalle 2014 ILT Exam Q49

Question Cognitive Level:
Memory or Fundamental Knowledge ___
Comprehension or Analysis _2_

10 CFR Part 55 Content:
55.41 _7_
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	263000 K5.01	
	Importance Rating	2.6	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to DC electrical distribution: Hydrogen generation during battery charging

Proposed Question: **Question 50**

What plant design feature associated with the Unit 2 250 VDC Battery ensures that the concentration of hydrogen created during the battery charging process remains BELOW explosive limits?

A. Battery Charger, 2DC03E limits DC current output to 125% of rated level

B. Battery room ventilation ensures at least 6 air changes per hour

C. Each battery cell has a spark arrestor mounted on top

D. Battery room ventilation maintains the battery temperature < 104°F

Proposed Answer: B

Explanation:

A) Incorrect. Plausible as this is an actual limit on the 2DC03E Battery Charger.

B) Correct. To purge the room of gaseous hydrogen liberated from the batteries at an air change rate of greater than 6 air changes per hour. This limits the hydrogen concentration to a level below the explosive limit.

C) Incorrect. Plausible as each battery cell has a spark arrestor which allows hydrogen to escape from the battery cell without the potential for a spark causing a flame to occur, but does not limit hydrogen concentration surrounding the battery.

D) Incorrect. Plausible as this is a limit the ventilation system maintains to protect the battery from overheating.

Technical Reference(s): 006, DC Distribution, (p. 21) Rev 10

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 006.00.05a; Recall the function, theory of operation, interlocks, trips and characteristics of the following DC Distribution System components and relate these items to overall system operation while operating the system or on an exam in accordance with the student text: a. Batteries.

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

Question History: Last NRC Exam NA

Question Cognitive Level:	Memory or Fundamental Knowledge	2
	Comprehension or Analysis	_

10 CFR Part 55 Content: 41.5, 45.3

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	264000 A1.09	
	Importance Rating	3.0	

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operating the emergency generators controls including: Maintaining minimum load on emergency generator (to prevent reverse power).

Proposed Question: **Question #51**

During the performance of LOS-DG-M2, "1A Diesel Generator Operability Test," the 1A EDG has just been paralleled to Bus 142Y and the following EDG indications are observed:

- 59.8 Hz
- 180 KW
- 200 KVAR
- 4.100 KV

What action is the crew required to perform?

A. Raise generator frequency to prevent generator lockout

B. Raise real load to prevent reverse power

C. Lower reactive load to prevent generator overcurrent

D. Lower voltage to prevent winding damage

Proposed Answer: B

Explanation:

Limitation D.8 of LOS-DG-M2 states when the EDG is paralleled, a minimum load of 200 kW must be maintained to prevent tripping on reverse power due to large load changes on the grid.

A) Incorrect. Plausible because the frequency is lower than the expected value of 60 Hz, but still within the band specified by the surveillance. A low generator frequency will cause the lockout relay to trip, but that value is approximately 57 Hz.
B) Correct. The minimum real load of the diesel is required to be 200 kW in order to prevent the EDG from tripping on reverse power.
C) Incorrect. Plausible because adjusting reactive load will change the current in the generator. A high reactive load will cause generator overcurrent. The given reactive load of 200 kVAR is higher than the real load, but not high enough to cause overcurrent.
D) Incorrect. Plausible because a high generator voltage will create excessive heat to cause generator winding damage. However, the given generator voltage is normal for the diesels at LaSalle and does not need to be adjusted in this situation.

Technical Reference(s): LOS-DG-M2 Rev. 104, page 8; 011 Emergency Diesel Generators pp. 65, 75, 79, 82-83

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 011.00.20, 011.05.20

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

Question History: Last NRC Exam Hope Creek 2005 ILT Exam (Q48)

Question Cognitive Level: Memory or Fundamental Knowledge 2
Comprehension or Analysis ___

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

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000 A2.01	
	Importance Rating	2.9	

K/A Statement: Instrument Air: Ability to predict the impacts of the following on the instrument air system and based on those predictions, use procedure to correct, control or mitigate the consequences of those abnormal operations: air dryer and filter malfunctions

Proposed Question: **Question 52**

Unit 1 is at 100% RTP

- LOA-IA-101, "Loss of Instrument/Service Air" has been entered due to an air dryer not operating properly

Per LOA-IA-101, the crew has secured control power to the Station Air Dryer.

What is the purpose of this action?

A. Isolates the air dryer

B. Places both Air Dryer Towers online

C. Maximizes the purge flow path

D. Switches to the standby drying tower

Proposed Answer: B

Explanation:

A) Incorrect. Plausible because as components which have failed are often isolated.

B) Correct. The given conditions indicate that there is a loss of station air and possible operating or valve issues with dryer 1SA02D. Turning control power off will deenergize the control solenoids, which should normally place both dryer chambers online and secure both purge inlet and purge exhaust flow paths [reference LOA-IA-101, step 4.1]. This should help recover station air header pressure.

C) Incorrect. Verifying purge air is maximized is required in annunciator procedure 1PM10J-B105 as an attempt to clear any moisture from the dryer. It is not required for the given conditions. Plausible because this is one of the responses for a problem with dryer flow on air dryer 1SA02D.

D) Incorrect. Locally verifying a switching failure as the cause of the alarm is required in annunciator procedure 1PM10J-B105 if a light on local panel on 1SA02D indicates a possible dryer cycle error. Plausible because this is the correct response for a problem with a switching error on air dryer 1SA02D.

Technical Reference(s):

- LOA-IA-101, Rev 14, Loss of Instrument/Service Air;
- Lesson Plan for System 120, Rev 12, Plant Air Systems;
- LOR-1PM10J-B105, Rev 5, U1 STA AIR DRYER Trouble;
- LOR-1PM10J-B206, Rev 2, RB Instr Air Press LO;

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 120.00.005: Recall the function, theory of operation, interlocks, trips, and characteristics of the following Plant Air Subsystem components and relate these items to overall system operation while operating the system or on an exam in accordance with the student text: c. Filters d. Dryers

Question Source:	Bank #	___	
	Modified Bank #	___	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	-	
	Comprehension or Analysis	3	
10 CFR Part 55 Content:	41.5 / 41.6		

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	400000 A3.01	
	Importance Rating	3.0	

K/A Statement: Ability to monitor automatic operations of the component cooling water systems including: Setpoints on instrument level signals for normal operations, warnings, and trips that are applicable to the CCWS.

Proposed Question: **Question #53**

Unit 1 was at 100% RTP when a small steam leak occurred inside the drywell.

- Drywell pressure is 2.0 psig and slowly RISING

The MOST limiting concern associated with the Reactor Recirculation System is the ____(1)____ because ____(2)____.

A. (1) RR Pump seal temperatures
   (2) RBCCW flow to the RR Pump seal coolers automatically isolated

B. (1) RR Pump seal temperatures
   (2) CRD Pump flow to the RR Pump seals automatically isolated

C. (1) RR Pump motor winding temperatures
   (2) RBCCW flow to the RR Pump motor winding coolers automatically isolated

D. (1) RR Pump motor winding temperatures
   (2) CRD Pump flow to the RR Pump motor winding coolers automatically isolated

Proposed Answer: C

Explanation:

RBCCW provides cooling to the RR Pump seal coolers, winding coolers, and bearing coolers. On high drywell pressure, RBCCW flow will automatically isolate to the containment. According to 114 RBCCW, the most limiting concern for loss of RBCCW flow to the RR Pumps are the motor winding temperatures.

A) Incorrect. Plausible because RBCCW flow is isolated to the RR Pumps on high drywell pressure and seal temperature will rise. Although the seal coolers lose flow, CRD pump flow is still available to the RR Pump seals, so seal temperature is not the most limiting concern.
B) Incorrect. Plausible as seal temperatures will rise due to RBCCW being automatically isolated to the RR Pump seal coolers. The second part of the distractor is not correct because CRD Pump flow is not automatically isolated to the RR Pump seals.
C) Correct. When RBCCW automatically isolates on high drywell pressure, the RR Pump motor winding seal coolers lose cooling, and this is the most limiting concern for the RR Pumps.
D) Incorrect. The first part of the distractor is correct. The second part is plausible because the CRD Pumps provide cooling flow to the RR Pump seals, however CRD Pumps do not automatically isolate and they do not provide flow to the RR Pump motor winding coolers.

Technical Reference(s): 022 Reactor Recirculation, page 39, 48; 114 RBCCW, pages 9 - 10

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 114.00.05d, 114.00.12, 114.05.12, 114.00.18e

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

Question History: Last NRC Exam Susquehanna 2011 ILT Exam Q18

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis 3

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

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201001 K6.02	
	Importance Rating	3.0	

K/A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the Control Rod Drive System: Condensate storage tanks

Proposed Question: **Question 54**

An earthquake results in damage to the Cycled Condensate Storage Tank (CST) causing level to lower.

What is the CST level (in feet) associated with the CRD pump trip?

A. 3

B. 5

C. 16

D. 17.5

Proposed Answer: B

Explanation:

A) Incorrect. Plausible as this is the level in the CST when the RCIC pump suction valves auto swap from the CST to the suppression pool.

B) Correct. The CRD pumps are supplied from the common suction header of the Cycle Condensate Storage Tank. If the tank level drops below 5 feet, the CRD pumps will lose suction pressure and trip.

C) Incorrect. Plausible as with the reactor operating the CST is to be maintained above 16 feet. This is also the level at which the CST low level alarm sounds.

D) Incorrect. Plausible as CST level is maintained by procedure at a minimum of 17.5 feet in accordance with LOP CY-01.

Technical Reference(s): 115, Condensate Transfer System, (p. 15) Rev 6; RPV Control; LOP CY-01, “Filling and Venting the Cycled Condensate Storage and Transfer System,” Rev 12;

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 115.00.18; Given various plant conditions, predict the response of the following supported systems to a loss of the Condensate Storage and Transfer System while operating the system, or on an exam in accordance with station procedures: o. Control Rod Drive System

Question Source:	Bank #	___	
	Modified Bank #	X	(Note changes or attach parent)
	New	___	
Question History:	Last NRC Exam	Brunswick 2016 NRC ILE (Q2)	
Question Cognitive Level:	Memory or Fundamental Knowledge		2
	Comprehension or Analysis		___
10 CFR Part 55 Content:	41.7, 45.7		

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201002 A4.05	
	Importance Rating	3.1	

K/A Statement: Ability to manually operate and/or monitor in the control room: Rod select matrix.

Proposed Question: **Question #55**

Unit 2 was operating at 95% RTP when a high worth control rod is withdrawn and the following annunciator is received:

- 1H13-P603-A406, RBM HI/INOP

Per LOR-1H13-P603-A406, what action is REQUIRED to reset the rod block?

A. BYPASS the Rod Worth Minimizer

B. Electrically disarm the affected control rod

C. Select an edge control rod, then select the control rod per the sequence

D. Attempt to insert the control rod to the previous sequence position

Proposed Answer: __C__

Explanation:

With RPIS in the inoperative condition, the control room loses all rod position indication. LOA-RM-101 states with RPIS inoperative, RCMS will not process notch insert (Insert command), notch withdraw (Withdraw command), or continuous withdraw. The only command to move rods is continuous insert.

A) Incorrect. Plausible as the RWM has rod withdrawal rod blocks. In this instance, the annunciator received is associated with the RBM system not the RWM.
B) Incorrect. Plausible as this is an action performed by a control rod which is not performing correctly (e.g. drifting).
C) Correct. Correct as the RBM system is not active when an edge rod is selected, and therefore serves as a means of resetting the RBM system.
D) Incorrect. Plausible as this would be the action for a control rod which lost rod position indication when withdrawn.

Technical Reference(s): LOR-1H13-P603-A406, RBM HI/INOP, Rev 6

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 047.00.05b

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 __2__
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 __7__
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201003 G2.4.45	
	Importance Rating	4.1	

K/A Statement: For Safety Function 1, (Control Rod and Drive Mechanism): Ability to prioritize and interpret the significance of each annunciator or alarm.

Proposed Question: **Question 56**

Unit 1 was at 100% RTP with the 1B Standby Liquid Control Pump Out of Service for maintenance:

- An automatic reactor SCRAM occurred resulting in a hydraulic ATWS
- Multiple control rods failed to insert
- Reactor power is 10%
- The crew is taking the actions of LGA-010, "Failure to SCRAM"
- ATWS choreography is complete including communications

The following annunciators (amongst others) are LIT:

- 1H13-P603-A305, 1A SBLC PMP AUTO TRIP
- 1H13-P602-A302, DIV 1 RR PMPS TRIP-ATWS INITIATED
- 1H13-P601-E105, REACTOR VESSEL LO WTR LVL 3 CONFIRMED
- 1H13-P603-A201, SCRAM DSCH SOUTH VOL/LVL HI

Which annunciator has the HIGHEST priority to REPORT to the Unit Supervisor?

A) 1H13-P602-A302, DIV 1 RR PMPS TRIP-ATWS INITIATED

B) 1H13-P601-E105, REACTOR VESSEL LO WTR LVL 3 CONFIRMED

C) 1H13-P603-A201, SCRAM DSCH SOUTH VOL/LVL HI

D) 1H13-P603-A305, 1A SBLC PMP AUTO TRIP

Proposed Answer: D

Explanation:

A) Incorrect. This should be an expected alarm for the given conditions. Plausible as an ATWS condition is occurring and reactor level would be lowered to at least -60 inches per LGA-010.

B) Incorrect. Depending upon the transient which resulted in the Scram, Level 3 could be an expected condition. Plausible because rapidly lowering RPV water level to at least -60 inches is required in LGA-010 if reactor power is above 3%.

C) Incorrect. Given the partially completed Scram in this question a high SDV would be an expected condition. Plausible because if there is a hydraulic ATWS the candidate would be expecting this annunciator.

D) Correct. With 1B SBLC Pump OOS and 1A SBLC failing to operate LGA-010 requires that RWCU be used to inject boron. With the other annunciators being expected conditions this annunciator is the highest priority to report to the Unit Supervisor.

Technical Reference(s): LOR-1H13-P603-A305, Standby Liquid Control Pump 1A Automatic Trip; LOR-1H13-P602-A302, Div 1 Reactor Recirculation Pumps Trip-Anticipated Transient Without Scram Initiated; LOR-1H13-P601-E105, Reactor Vessel Low Water Level Three Confirmed initiated; LOR-1H13-P603-A201, Scram Discharge South Volume/Level High initiated; LGA-010, Rev 18, Failure to Scram

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 500.01: During performance of plant tasks, demonstrate the following human performance behaviors IAW the identified procedures: Annunciator Response

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

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge __
Comprehension or Analysis 2

10 CFR Part 55 Content: 41.10 / 43.5 / 45.3 / 45.12

Comments: **This is a KA match as the question requires the applicant to assess various annunciators and determine their priority based on plant conditions associated with control rods failing to insert with a reactor scram required.**

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	204000 K1.15	
	Importance Rating	3.1	

K/A Statement: Knowledge of the physical connections and/or cause-effect relationships between the reactor water cleanup system and the following: Leak Detection.

Proposed Question: **Question #57**

Unit 1 was operating at 100% RTP

- ‘A’ RWCU Pump is running

Subsequently,

- 1H13-P601-C411, DIV 1 RWCU ∆ FLOW HI, is LIT

IMMEDIATELY after receiving this alarm, 1G33-F004, RWCU Outboard Isolation Valve, will be ___(1)___ and the ‘A’ RWCU Pump will be ___(2)___.

A. (1) closed
(2) tripped

B. (1) closed
(2) running

C. (1) open
(2) tripped

D. (1) open
(2) running

Proposed Answer: D

Explanation:

The Leak Detection System will provide an alarm to the control room on high differential flow between the inlet and return flow rates on the RWCU system. High differential flow is indicative of a leak in the system. After receiving this alarm, the system will isolate in 45 seconds if the condition has not been corrected by closing the outboard isolation valve (F004) which causes the running pump to trip. The inboard isolation valve (F001) will remain open (this valve closes on Div 2 High Differential Flow).

A) Incorrect. Plausible because the Leak Detection System will immediately isolate the system from other alarms, such as high differential temperature. However, for high differential flow, the crew has 45 seconds to correct the condition before the system isolates.
B) Incorrect. Plausible because Leak Detection System will immediately isolate the system from other alarms, such as high differential temperature. However, for high differential flow, the crew has 45 seconds to correct the condition before the system isolates. The second part of the distractor is correct, because the pump is still running, and this is plausible because the alarm manual only states the automatic action is for the valve to close. However, the pump also automatically trips when the valve is closed (027, p. 15)
C) Incorrect. Plausible if the applicant believes an RWCU isolation occurs immediately with a high differential flow condition AND with the inboard valve (F001) closing on Div. 1 leak detection high differential flow. This is also plausible if the applicant believes only the pump trips on system isolation.
D) Correct. With the Div 1 High Differential Flow alarm, the crew has 45 seconds before the RWCU system isolates (F004) and the running RWCU pump trips. Immediately after the alarm is received, the outboard valve is still open and the pump is still running.

Technical Reference(s): 027 RWCU, pp 20, 28, 31; LOR-H13-P601-C411 Rev 007; RT-1

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 027.00.16c

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

Question History: Last NRC Exam Hatch 2013 ILT NRC Exam (Q3)

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis 3

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

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	214000 K3.01	
	Importance Rating	3.0	

K/A Statement: Knowledge of the effect that a loss or malfunction of the rod position information will have on the following: RWM.

Proposed Question: **Question 58**

- Unit 1 is at 100% RTP
- Control Rod 42-59 loses position indication
- The Rod Worth Minimizer Mode Select Switch is in the NORMAL position

The loss of control rod position indication information for Control Rod 42-59 causes the Rod Worth Minimizer to _________________.

A. initiate a WITHDRAW rod block ONLY

B. initiate an INSERT and WITHDRAW rod block

C. be unaffected as reactor power is above the Low Power Set Point

D. be unaffected as this is a peripheral control rod

Proposed Answer: B

Explanation:

A) Incorrect. Plausible as the RWM does initiate withdraw only blocks if a rod is out of sequence by a certain criteria. In this instance, the rod position is lost so a rod withdraw block and insert block is in effect.

B) Correct. The RWM takes rod position and compares it to sequence steps. In this case, with a loss of RPI, a withdraw and insert block would occur. This is still the case with power above 10% (LPSP) as the RWM is set to full power blocks.

C) Incorrect. Plausible as RWM rod blocks can be set to be bypassed above the LPSP of 10% steam flow. LaSalle maintains rod blocks in effect above the LPSP up to 100% RTP.

D) Incorrect. Plausible as the rod block monitoring circuit does not initiate rod blocks for peripheral control rod movements. The student could confuse a RBM initiated rod block with a RWM initiated rod block.

Technical Reference(s):

- Lesson Plan 048, Rev 3, Rod Worth Minimizer
- LOA-RM-101, Rev 23, Unit 1 RCMS Abnormal Situations
- LOR-1PM01J-A111, Rev 5, UPS Trouble

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 048.00.021: Given various plant conditions, predict how the Rod Worth Minimizer System will respond to various system component failures while operating the system or on an exam in accordance with student text.

Question Source:	Bank #	___	
	Modified Bank #	___ (Note changes or attach parent)	
	New	X	
Question History:	Last NRC Exam	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge		2
	Comprehension or Analysis		___
10 CFR Part 55 Content:	41.7 / 45.4		

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	215001 K4.01	
	Importance Rating	3.4	

K/A Statement: Knowledge of traversing in-core probe design feature(s) and/or interlocks which provide for the following: Primary containment isolation.

Proposed Question: **Question #59**

During hand cranking of the TIPs IAW LOP-NR-06, "Traversing Incore Probe (TIP) Operation" on Unit 2, the following actions and indications occurred:

- Channel 'A' Drive Control Mode Switch is turned to 'OFF'
- On Drive Control Unit panel 2H13-P607, the In-Shield light is NOT LIT

Based on the above actions and indications, which of the following conditions exist?

A. Group 2 isolation logic is inoperable

B. TIP automatic isolation function is inoperable

C. Detectors are fully retracted in the shield chamber

D. Operators will NOT be able to activate the shear valves

Proposed Answer: B

Explanation:

Precaution C.4 of LOP-NR-06 states that the Drive Control Mode Switch should not be turned off when the detector is not in the in-shield position. Doing so will make the automatic isolation function of the TIP system inoperable. The precaution also states the Group 7 initiation logic is not affected.

A) Incorrect. Plausible because the automatic isolation of the TIP system is inoperable, which is a part of the Group 2 isolation.
B) Correct. With the in-shield light off, the TIP detectors are not fully retracted. Precaution C.4 states when the switch is off and the detectors are not in the shield chamber, the TIP automatic isolation function is inoperable.
C) Incorrect. Plausible because the in-shield light not being lit may be considered to indicate that an off-normal or incorrect position has been cleared, similar to when APRM/LPRM downscale lights are lit. When the downscale lights are lit, then an undesired condition exists; however, with the in-shield light NOT lit, an undesired condition exists in this situation.
D) Incorrect. Plausible because in this situation, the TIP detectors are not fully retracted, and the TIP ball valves are unable to be closed (046 p. 4). Therefore, it is reasonable to believe the shear valves will also be unable to isolate, but these valves can be operated as an emergency means to isolate the TIP guide tube in the event the detectors do not retract.

Technical Reference(s): 046 Traveling In-core Probes, pp. 4 and 14; LOP-NR-06, Rev. 032, p. 5

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 046.00.16a

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

Question History: Last NRC Exam N/A – LaSalle ILT Bank

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis 2

10 CFR Part 55 Content: 55.41 7
55.43 ___

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	219000 K2.02	
	Importance Rating	3.1	

K/A Statement: Knowledge of electrical power supplies to the following: Pumps (Torus/Suppression Pool Cooling Mode)

Proposed Question: **Question 60**

Which of the following bus losses will affect the ‘1A’ RHR Pump running in “Suppression Pool Cooling Mode?”

A. 141X

B. 142X

C. 141Y

D. 142Y

Proposed Answer: C

Explanation:

A) Incorrect. Plausible as 141X powers Division 1 non-ESF 4160VAC loads.

B) Incorrect. Plausible as 142X powers 4160VAC loads.

C) Correct. The '1A' RHR pump is powered by Bus 141Y.

D) Incorrect. Plausible as 142Y powers Division 2 ESF 4160VAC loads.

Technical Reference(s):

- Lesson Plan 064, Residual Heat Removal System (p. 42) Rev 17
- Training Drawing, Rev 2, Residual Heat Removal System, RH-1

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 064.00.05: Recall the function, theory of operation, interlocks, trips, and characteristics of the following Residual Heat Removal System components and relate these items to overall system operation while operating the system or on an exam in accordance with the student text: a. Pumps

Question Source:	Bank #	___	
	Modified Bank #	___	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	2	
	Comprehension or Analysis	___	
10 CFR Part 55 Content:	41.7		

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	259001 K5.02	
	Importance Rating	2.5	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to reactor feedwater system: Water hammer.

Proposed Question: **Question #61**

On a loss of the condensate system at normal operating pressures and temperatures, the crew should NOT restart the condensate system until the feedwater system has had sufficient time to cool down, or the feedwater system is isolated by the feedwater pump suction valves and the feedwater seal injection system, in order to prevent ____________.

A. high water level trips in the HP FW heaters

B. water hammer within the condensate system

C. pump run-out conditions in the condensate pump

D. windmilling a TDRFP during condensate pump restart

Proposed Answer: B

Explanation:

The operational implications of restarting a condensate pump with the feed system at normal temperature and pressure, or if the feed suction is isolated, will cause water hammer in the condensate system, unless the feed system has had sufficient time to cool down. Adhering to precaution C.10 in LOP-CD-03 will prevent water hammer.

A) Incorrect. Plausible because high water level trips can be a concern with cooler water running through the heater with increased condensation but extraction steam at this point would be isolated to the heaters due to lack of system flow, so this is not the reason for the precaution.
B) Correct. Precaution C.10 in LOP-CD-03 states the reason is to prevent water hammer in the condensate system.
C) Incorrect. Plausible because pump runout can be a concern with this system but the system is started with Feedwater valves closed and the pumps running on their min flow valves. This is NOT an issue when first starting the system.
D) Incorrect. Plausible because it is based on it is a precaution step from LOP-CD-03 "STARTUP AND OPERATION OF THE CONDENSATE AND CONDENSATE BOOSTER SYSTEM" windmilling a TDRFP is possible when opening the min flow valve on the associated pump.

Technical Reference(s): LOP-CD-03 Rev. 036, p. 4 - 5

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 075.05.20

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

Question History: Last NRC Exam LaSalle 2014 ILT NRC Exam (Q60)

Question Cognitive Level: Memory or Fundamental Knowledge 2
Comprehension or Analysis ___

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

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	268000 A1.01	
	Importance Rating	2.7	

K/A Statement: Radwaste: Ability to predict and/or monitor changes in parameters associated with operating the Radwaste controls including: Radiation Levels

Proposed Question: **Question 62**

Unit 1 was at 100% RTP when indications of a fuel defect were identified.

Subsequently, a reactor SCRAM occurs requiring Reactor Water Cleanup (RT) rejection to the Radwaste system.

Radiation levels will be higher near the ______________ due to the RT reject.

A. 0WX01T, Waste Sludge Tank

B. 1WE02T, Waste Collector Surge Tank

C. 0WX02T, URC Sludge Tank

D. 1WE05T, Radwaste Discharge Tank

Proposed Answer: B

Explanation:

A) Incorrect. Plausible as the Waste Sludge Tank receives water from the fuel pool cooling system filter / Demineralizer backwashes and the chemical waste collector tank bottom blows, among others.

B) Correct. There is a specific notation in Lesson Plan 121, Liquid Processing, Page 8 regarding the possibility of a sudden in-surge of RT Reject water into the waste collector surge tanks post SCRAM.

C) Incorrect. Plausible as the URC Sludge Tank receives water from Waste Flocculator Tank Bottom Blows, and the Radwaste Building 734 floor drains among others but does not receive water from RT Reject.

D) Incorrect. Plausible as the 1WE05T is in the radwaste system and holds water prior to offsite release.

Technical Reference(s):

- Lesson Plan 121, Liquid Processing and Sumps, (p. 8) Rev 8
- Lesson Plan 122, Waste Processing, Rev 6

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 121.00.05a2: Recall the function, theory of operation, interlocks, trips, and characteristics of the following Liquid Processing and Sumps components and relate these items to overall system operation while operating the system or on an exam in accordance with the student text:

a. Equipment Drain System
  1. Waste Collector Tanks
  2. Waste Collector Surge Tanks
  3. Waste Flocculating Tank
  6. Waste Sample Tanks

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

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge 3
Comprehension or Analysis __

10 CFR Part 55 Content: 41.5 / 45.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	286000 A2.12	
	Importance Rating	3.1	

K/A Statement: Ability to (a) predict the impacts of the following on the fire protection system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low diesel fuel supply.

Proposed Question: **Question #63**

The ‘0A’ Diesel Fire Pump was RUNNING per LOS-FP-M6, “Diesel Fire Pump Operational Check.”

Subsequently,

- 1PM10J-B501, 0A DIESEL FIRE PMP DAY TANK LVL HI-HI LO, is LIT
- The Equipment Operator reports that 0LIS-DO007 for the ‘0A’ Diesel Fire Pump Day Tank is at the low level alarm setpoint

(1) What is the status of the ‘0A’ Diesel Fire Pump?
(2) What action MUST be taken IAW LOP-DO-03, “Transferring Oil to the Diesel Fire Pump Day Tanks”?

A. (1) Running
(2) Stop the Diesel Fire Pump Fuel Oil Transfer Pump

B. (1) Running
(2) Fill the ‘0A’ Diesel Fire Pump Day Tank within band

C. (1) Tripped
(2) Stop the Diesel Fire Pump Fuel Oil Transfer Pump

D. (1) Tripped
(2) Fill the ‘0A’ Diesel Fire Pump Day Tank within band

Proposed Answer: B

Explanation:

During the performance of LOS-FP-M6, the diesel fire pumps are operated for 60 minutes, which will use diesel fuel. The time of one hour after the completion of the STP indicates that there is a fuel oil leak. The alarm received in the control room is a common high and low level alarm, so the applicant must know the provided indication is low and not high. The minimum required fuel level in the day tank is 19” (LOP-DO-03) and the indicated level is 20.5.” Therefore, the diesel fire pump remains operable even though the low level alarm is received.

A) Incorrect. Plausible because the low level alarm gives operators time to respond before the diesel fire pump would trip due to a loss of fuel inventory. The second part is plausible because this is the required action if the level in the day tank is high.
B) Correct. The minimum required level per the TRM is 170 gal, which corresponds to 19.” An indication of 20.5” is still enough for the diesel fire pump to be running. The alarm manual directs operators to fill the day tank level to a level band of 27” to 29” IAW LOP-DO-03.
C) Incorrect. Plausible because the diesel fire pump would eventually trip on low fuel oil day tank level, which may be understood to occur when the alarm condition exists. However, the low level alarm at 21.5” gives operators time to respond before the diesel fire pump trips on low fuel level. The second part is plausible because this is the required action if the level in the day tank is high.
D) Incorrect. Plausible as the second part of the distractor is correct. Also because the diesel fire pump would eventually trip on low fuel oil day tank level, which may be understood to occur when the alarm condition exists. However, the low level alarm at 21.5” gives operators time to respond before the diesel fire pump trips on low fuel level.

Technical Reference(s): 125 Fire Protection, p. 45; LOP-DO-03 Rev. 023, pp. 3-5; LOS-FP-M6 Rev. 017, pp. 8-18; LOR-1PM10J-B501 Rev. 004; TRM 3.7.j Rev. 003

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 125.00.06a4, 125.00.22

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

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis 3

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

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	290001 A3.01	
	Importance Rating	3.9	

K/A Statement: Ability to monitor automatic operations of the secondary containment including: Secondary Containment Isolation

Proposed Question: **Question 64**

Unit 2 experienced a slow leak from the reactor recirculation system resulting in Drywell pressure rising to 2 psig.

Which condition reflects the status of the Reactor Building Ventilation Dampers (VR)?

A. Unit 2 VR Supply and Exhaust Dampers are CLOSED ONLY

B. BOTH Unit 1 AND Unit 2 Exhaust Dampers are CLOSED ONLY

C. BOTH Unit 1 AND Unit 2 VR Supply and Exhaust Dampers are CLOSED

D. BOTH Unit 1 AND Unit 2 Supply Dampers are CLOSED ONLY

Proposed Answer: C

Explanation:

A) Incorrect. Plausible as this is the Unit that is directly affected by the condition. A high pressure in the main steam tunnel on Unit 2 would have resulted in this condition.

B) Incorrect. Plausible as closing the exhaust dampers could be seen as stopping the release of radioactive material from the reactor building ventilation stack. These dampers would close, but so would the inlet dampers.

C) Correct. With a common reactor building secondary containment, both units' inlet and exhaust VR dampers close on a high drywell pressure signal.

D) Incorrect. Plausible as closing only the Supply dampers would result in a strong vacuum in the RB which is normally how contamination is prevented from escaping secondary containment.

Technical Reference(s):

- Lesson Plan 118, Reactor Building HVAC (p. 20) Rev 13
- Training Drawing VR-1, Rev 2, Reactor Building Ventilation Systems

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 118.00.05: Recall the function, theory of operation, interlocks, trips, and characteristics of the following VR System components and relate these items to overall system operation while operating the system or on an exam in accordance with the student text: b. Isolation Dampers 4 YA(B) and 5 YA(B)

Question Source:	Bank #	___	
	Modified Bank #	___	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	2	
	Comprehension or Analysis	___	
10 CFR Part 55 Content:	41.7 / 45.7		

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	290002 K4.05	
	Importance Rating	3.3	

K/A Statement: Knowledge of reactor vessel and internals design feature(s) and/or interlocks which provide for the following: Natural circulation.

Proposed Question: **Question #65**

At 0700, Unit 2 was in Mode 4 with the following conditions:

- Both Reactor Recirculation pumps are secured
- ‘B’ RHR Loop is in Shutdown Cooling Mode with a flowrate of 6500 gpm
- 2B21-R605, RX LVL SHUTDOWN RG indicates +40 inches

At 0701, the following alarm is received:

- 2H13-P601-B106, 2B RHR PMP AUTO TRIP, is LIT

Which action, if any, is the crew required to perform to promote NATURAL CIRCULATION through the reactor core?

A. Maximize RWCU Bottom Head Drain flow

B. Re-establish shutdown cooling flow with ‘A’ RHR Loop

C. Raise RPV level to greater than +50 inches

D. No action is required, natural circulation exists in the core

Proposed Answer: C

Explanation:

In LOP-RH-07, Precaution C.16 states when entering cold shutdown, RPV level should be above the bottom of the pre-dryers on the steam separators in order to promote natural circulation in the event that forced circulation is lost. This corresponds to an RPV level of above +50” as indicated on the Shutdown Range.

A) Incorrect. Plausible because a loss of forced circulation may result in thermal stratification in the bottom vessel head. Raising bottom head drain flow using RWCU will reduce thermal stratification but it will not promote natural circulation.
B) Incorrect. Plausible because the crew will take steps to recover shutdown cooling. Using the ‘A’ RHR loop will re-establish forced circulation and not promote natural circulation.
C) Correct. Raising RPV level to greater than 50” will promote natural circulation.
D) Incorrect. Plausible if the applicant believes the RPV level in this situation is adequate to promote natural circulation. However, this level is too low for natural circulation to occur.

Technical Reference(s): 020 Reactor Vessel & Internals, pp. 6, 37-38; LOP-RH-07 Rev. 080, p. 6; NB-5

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 020.00.04a

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

Question History: Last NRC Exam Limerick/Peach Bottom 2005 LSRO Exam (Q47)

Question Cognitive Level: Memory or Fundamental Knowledge ___
Comprehension or Analysis 3

10 CFR Part 55 Content: 55.41 7
55.43 ____

Comments:

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.1.38	
Importance Rating	3.7	

K/A Statement: Knowledge of the station's requirements for verbal communications when implementing procedures.

Proposed Question: **Question 66**

Unit 2 is at 100% RTP when the following occurred:

- 2H13-P603-B501, PRI CNMT PRESSURE HI/LO, is LIT
- Drywell Pressure indicates 0.60 psig and is trending UP SLOWLY

After the alarm is reported, which is the correct way for the Operator At-the Controls to communicate the Drywell pressure condition to the crew?

A. "Attention for a brief, Drywell Pressure is 0.60 psig and rising slowly, end of brief."

B. "Attention for a brief, Drywell Pressure is 0.60 psig and increasing slowly, end of brief."

C. "Attention for an update, Drywell Pressure is 0.60 psig and rising slowly, end of update."

D. "Attention for an update, Drywell Pressure is 0.60 psig and increasing slowly, end of update."

Proposed Answer: C

Explanation:

A) Incorrect. A change in a plant parameter needing to be communicated verbally to the rest of the crew is an update not a brief. See Lesson Plan 500, Section II.K. The use of the word rising is correct for the given conditions. Plausible as the use of the words "brief" and "rising" is routine.

B) Incorrect. A change in a plant parameter needing to be communicated verbally to the rest of the crew is an update not a brief. The use of the word "increase" is specifically listed as needing to be avoided in HU-AA-101, step 4.4.4. Plausible as the use of the words "brief" is routine.

C) Correct. Update is correct for the given conditions (major change in plant status) and rising is to be used over the word increase.

D) Incorrect. Update is properly used but increase is to be avoided. Plausible as the word increase is used for other purposes in the control room and is routine.

Technical Reference(s):

- Lesson Plan 500, Rev 1, Control Room Techniques and Good Operating Practices
- HU-AA-101, Rev 10, Human Performance Tools and Verification Practices
- LOR-1H13-P603-B501, Rev 4, Primary Containment Pressure Hi/Lo
- OP-AA-104-101, "Communications," (p. 3) Rev 4

Proposed references to be provided to applicants during examination: N/A

Learning Objective:

Question Source:	Bank #	X	
	Modified Bank #	___ (Attached parent)	
	New	____	
Question History:	Last NRC Exam	Oyster Creek 2009 NRC Exam (Q74)	
Question Cognitive Level:	Memory or Fundamental Knowledge		2
	Comprehension or Analysis		___
10 CFR Part 55 Content:	41.10 / 45.13		

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.1.44	
	Importance Rating	3.9	

K/A Statement: Knowledge of RO duties in the control room during fuel handing such as responding to alarms from the fuel handling area, communication with the fuel storage facility, systems operated from the control room in support of refueling operations, and supporting instrumentation.

Proposed Question: **Question #67**

Fuel handlers are loading a new fuel assembly into the core. When the fuel assembly is halfway into the core, direct communication is lost between the control room and the refueling bridge. Which of the following actions is required regarding refueling operations?

A. Continue fuel moves after verifying proper operation of the SRMs

B. Continue fuel moves after obtaining permission from the Unit Supervisor

C. Suspend fuel moves after evacuating the Refuel Floor

D. Suspend fuel moves after moving the fuel bundle to a safe location

Proposed Answer: D

Explanation:

During refueling operations, the RO in the control room coordinates the conduct of refueling activities and monitors nuclear instrumentation (OP-AA-300). Part of coordinating refueling activities is to ensure direct communication is maintained between the control room and the refueling machine. TRM 3.9.b states that direct communication shall be maintained between the control room and refueling machine personnel. When direct communication is lost, fuel moves must be suspended immediately.

A) Incorrect. Plausible because it is an RO duty to monitor for potential criticality during refueling operations in the reactor by monitoring the SRM detectors.
B) Incorrect. Plausible because it is reasonable to believe fuel moves can continue when the refuel machine is not disabled, a fuel bundle is not damaged, and no radiation alarms are present. The Unit Supervisor or SRO directly supervise core alterations and give permission to move fuel. However, with a loss of communication, refueling can only commence once direct communication has been re-established and the surveillance to demonstrate direct communication exists has been completed.
C) Incorrect. Plausible because the TRM states refueling activities must be suspended, but it is only required to evacuate the refuel floor if there is indication of fuel damage, such as a high radiation alarm.
D) Correct. TRM 3.9.b states when direct communication is lost to suspend all fuel movements, and the basis states to place the fuel in a safe location.

Technical Reference(s): LOA-AR-101 Rev 5 (p. 4); OP-AA-300 Rev 13 (pp. 5-7); TRM 3.9.b Rev. 001; TRM B3.9.b Rev. 001

Proposed references to be provided to applicants during examination: N/A

Learning Objective: ___

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

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge 2
Comprehension or Analysis ___

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

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.2.14	
	Importance Rating	3.9	

K/A Statement: Knowledge of the process for controlling equipment configuration or status.

Proposed Question: **Question 68**

LOS-DG-M3, "1B Diesel Generator Operability Test," is being performed with step 4.14.5.2 to be performed NEXT:

4.14.5.2 UNLOCK and OPEN 1DO018, 1B DG Day Tank Drain Valve (1B DG Day Tank Room), and OBSERVE flow through sightglass.

4.14.5.3 DRAIN tank for one (1) minute.

4.14.5.4 CLOSE and LOCK 1DO018, 1B DG Day Tank Drain Valve (1B DG Day Tank Room). (Tech Spec 3.8.1 and 3.8.2)

______/______

______/______

The change in valve position is ___(1)___ to be recorded in the Locked Valve Log. Shift Management permission to UNLOCK the valve ___(2)___ required on the Work Order.

A. (1) required
(2) was

B. (1) NOT required
(2) was

C. (1) NOT required
(2) was NOT

D. (1) required
(2) was NOT

Proposed Answer: B

Explanation:

A) Incorrect. Licensee procedures state that locked valves can be controlled by certain approved procedures so the first part is wrong. The second part is correct. Plausible as the control of locked valves is usually documented in the Locked Valve Log and the second part is correct.

B) Correct. Licensee procedure OP-AA-108-103, step 4.1 states that log entries are not required if the equipment is controlled by approved procedures that include documentation of restorations and verification practices. For tests and clearances, authorization is granted to move a locked valve when shift management releases the test or clearance.

C) Incorrect. The first part is correct and the second part is incorrect. Plausible as the control of locked valves is well maintained and it would be plausible for shift management to have additional control.

D) Incorrect. Licensee procedures state that locked valves can be controlled by certain approved procedures so the first part is wrong. The second part is also incorrect. Plausible as the control of locked valves is usually documented in the Locked Valve Log and would be expected by the candidate.

Technical Reference(s): OP-AA-108-103, "Locked Equipment Program" (Step 4.1) Rev 2

Proposed references to be provided to applicants during examination: N/A

Learning Objective:

Question Source:	Bank #	___	
	Modified Bank #	___	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	2	
	Comprehension or Analysis	__	
10 CFR Part 55 Content:	41.10 / 43.3 / 45.13		

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.2.22	
	Importance Rating	4.0	

K/A Statement: Knowledge of limiting conditions for operations and safety limits.

Proposed Question: **Question #69**

If a Safety Limit (SL) is violated, the crew is required to ___(1)___ within ___(2)___.

A. (1) restore compliance with all SLs ONLY
(2) one hour

B. (1) restore compliance with all SLs ONLY
(2) two hours

C. (1) restore compliance with all SLs AND insert all insertable control rods
(2) one hour

D. (1) restore compliance with all SLs AND insert all insertable control rods
(2) two hours

Proposed Answer: D

Explanation:

If a Safety Limit is violated, the crew is required to restore compliance with all safety limits and insert all insertable control rods within 2 hours.

A) Incorrect. Plausible because the crew is required to restore compliance with all SLs, but they are also required to insert all insertable control rods. The second part is plausible because operators are required to take prompt action to restore SLs to prevent fuel damage, but the TS allows for 2 hours.
B) Incorrect. Plausible because the crew is required to restore compliance with all SLs, but they are also required to insert all insertable control rods. The second part is correct.
C) Incorrect. Plausible because the first part is correct as the crew is required to restore compliance with all SLs and insert all insertable control rods, but the second part is incorrect as they have 2 hours to perform these actions.
D) Correct. The crew has 2 hours to restore compliance to all SLs and insert all insertable control rods.

Technical Reference(s): 2.0 Safety Limits

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 201.00.02

Question Source:	Bank #	____	
	Modified Bank #	____	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	N/A – LaSalle ILT Bank	
Question Cognitive Level:	Memory or Fundamental Knowledge	2	
	Comprehension or Analysis	____	
10 CFR Part 55 Content:	55.41	**5**	
	55.43	____	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.2.43	
	Importance Rating	3.0	

K/A Statement: Knowledge of the process used to track inoperable alarms.

Proposed Question: **Question 70**

Operational and maintenance practices are ineffective to eliminate a NUISANCE ALARM and it will remain in for GREATER than one shift.

Per OP-AA-103-102, "Watch Standing Practices," what is required to disable the nuisance alarm input?

(1) Obtain Unit Supervisor permission
(2) Initiate an Equipment Status Tag (EST)
(3) Make a note in the appropriate turnover
(4) Evaluate for increased monitoring for the disabled alarm function

A. (1), (2), and (4) ONLY

B. (1), (2), and (3) ONLY

C. (3) and (4) ONLY

D. (1), (2), (3), and (4)

Proposed Answer: D

Explanation:

A) Incorrect. Plausible as these three conditions are required per OP-AA-103-102.

B) Incorrect. Plausible as these three conditions are required per OP-AA-103-102.

C) Incorrect. Plausible as these two conditions are required per OP-AA-103-102.

D) Correct. Licensee procedure OP-AA-103-102, step 4.5.5 states that the four items noted above are required to be performed if an alarm is to be disabled for more than one shift.

Technical Reference(s):

- OP-AA-103-102, Watch Standing Practices (p. 10) Rev 18
- OP-AA-108-105, Rev 11, Equipment Deficiency Identification and Documentation

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 762.02.00 - During performance of tasks demonstrate the following performance behaviors: Annunciator Response, IAW OP-AA-103-102.

Question Source:	Bank #	___	
	Modified Bank #	___	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	2	
	Comprehension or Analysis	___	
10 CFR Part 55 Content:	41.10 / 43.5 / 45.13		

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.3.5	
	Importance Rating	2.9	

K/A Statement: Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: **Question #71**

1H13-P601-B209, Common Area Monitoring Radiation High is LIT.

20 SECONDS later the back panel is checked:

- 1H13-P624 shows that ARM channel 3-9, Rx Bldg Trackway "Hi" alarm is LIT
- Rx Bldg Trackway meter displays NORMAL radiation levels for the area

The Reactor Operator will ___(1)___ the alarm on 1H13-P624 AND inform ___(2)___.

A. (1) isolate
(2) the Unit Supervisor

B. (1) isolate
(2) Instrument Maintenance

C. (1) reset
(2) the Unit Supervisor

D. (1) reset
(2) Instrument Maintenance

Proposed Answer: C

Explanation:

A) Incorrect. Plausible if the applicant believes the alarm is not functional based on the conditions given, it would be isolated IAW LOP-AR-01. The Unit Supervisor is the correct individual to notify based on the conditions given.
B) Incorrect. Plausible if the applicant believes the alarm is not functional based on the conditions given, it would be isolated IAW LOP-AR-01 and IMD would be informed to make repairs.
C) Correct. The question stem indicates a condition of a spurious high rad condition which has cleared. The applicant would be expected to reset the alarm and inform the Unit Supervisor that the condition occurred and has been reset.
D) Incorrect. Plausible as resetting the alarm is correct and if the applicant believes that the conditions presented indicate a deficiency with the alarm then IMD would be notified to make repairs.

Technical Reference(s): 051 Area Radiation Monitoring Rev 11 (p. 12); LOR-H13-P601-B301 Rev1; LOP-AR-01 Rev 11

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 051.00.05

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

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis 2

10 CFR Part 55 Content: 55.41 **11-12**
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.3.12	
	Importance Rating	3.2	

K/A Statement: Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question: **Question 72**

Unit 2 is at 5% RTP:

- The Reactor Mode Switch is in STARTUP
- A Drywell entry is in progress with personnel currently investigating a suspected leak

Subsequently,

- A high worth Control Rod begins to drift OUT

The crew will...

A. Scram Unit 2 AND evacuate the Drywell

B. Scram Unit 2 AND continue to investigate the leak

C. evacuate the Drywell AND follow-up with actions to insert ONLY the drifting control rod

D. continue to investigate the leak AND follow-up with actions to insert ONLY the drifting control rod.

Proposed Answer: A

Explanation:

A) Correct. Per LAP-900-45, with personnel inside the drywell at power, the Reactor Operator SHALL scram the reactor immediately upon observing any unexpected power increase AND evacuate all personnel from the Drywell.

B) Incorrect. While it is correct to scram the unit following a power spike, the personnel in the drywell are required to evacuate. However, this is plausible because following the power spike the rad levels should lower due to the lowering of reactor power from the scram.

C) Incorrect. In this situation a SCRAM is required. Plausible because you are supposed to evacuate the drywell. This is also plausible because during normal operations the crew would insert the drifting control rod per LOA-RD-101.

D) Incorrect. In this situation a SCRAM is required. Plausible because per LAP-900-45, certain areas of the drywell are accessible up to 40% reactor power. This is also plausible because during normal operations the crew would insert the drifting control rod per LOA-RD-101.

Technical Reference(s): LAP-900-45,

Proposed references to be provided to applicants during examination: N/A

Learning Objective:

Question Source:	Bank #	X	
	Modified Bank #	_	(Note changes or attach parent)
	New	___	
Question History:	Last NRC Exam	LaSalle 2018 NRC ILE (Q73)	
Question Cognitive Level:	Memory or Fundamental Knowledge	_	
	Comprehension or Analysis	2	
10 CFR Part 55 Content:	41.12 / 45.9 / 45.10		

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.4.14	
	Importance Rating	3.8	

K/A Statement: Knowledge of general guidelines for EOP usage.

Proposed Question: **Question #73**

An expected entry condition is met for an LGA due to placing the plant in a configuration needed to support a surveillance.

Per the LPGP-PSTG-01S01, Plant Specific Technical Guidelines Section 1 - Introduction, which of the following responses of the crew to reaching the expected LGA entry condition is allowed?

A. Enter the LGA and perform actions to restore the parameters

B. Enter the LGA, then the LGA may be exited because an emergency does not exist

C. Stop the surveillance and obtain permission from the Shift Manager to continue

D. Continue with the surveillance as no further action is required

Proposed Answer: B

Explanation:

When the performance of a surveillance causes plant conditions to meet an entry condition for an LGA, the BWROG gives two acceptable responses as described in Rule #1 of General Rules for Flowchart Use (p. 20). The first step is to enter the LGA then the LGA may be exited based on the fact that an emergency condition does not exist. The second is not to enter the LGA and announce that this is an expected condition and to also make a log entry of the entry condition that was met and stating no emergency exists.

A) Incorrect. Plausible because this is the expected response when an emergency condition exists. In this situation, the performance of the surveillance is the cause of the entry condition being met, and no emergency exists.
B) Correct. One of the acceptable responses is to enter the LGA then exit because no emergency exists.
C) Incorrect. It is reasonable to stop the performance of an evolution when the entry condition to an LGA is met. In this situation, the entry condition is expected and the crew is not required to stop the surveillance.
D) Incorrect. Plausible because the crew will continue with the surveillance since this is an expected condition. However, the entry of the condition needs to be announced, and the crew is required to make a log entry stating which entry condition was met and state that no emergency exists

Technical Reference(s): LGA Flow Chart Use Rev 21 (p. 20)

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 400.00.01, 400.00.18

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

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge 2
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 **10**
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.4.37	
	Importance Rating	3.0	

K/A Statement: Knowledge of the lines of authority during implementation of the emergency plan.

Proposed Question: **Question 74**

A Site Area Emergency has been declared.

The Technical Support Center (TSC) has been activated and has assumed command and control.

Who has Command Authority?

A. TSC Director

B. Operations Manager

C. Station Emergency Director

D. Corporate Emergency Director

Proposed Answer: C

Explanation:

A) Incorrect. Plausible as the TSC Director is an actual TSC ERO position, but the TSC Director reports to the Station Emergency Director and is responsible for the content of information transmitted from the TSC to other facilities or agencies and for supporting overall TSC activities.

B) Incorrect. Plausible as the Operations Manager is an actual TSC ERO position, but the Operations Manager reports to the Station Emergency Director and determines the extent of station emergencies, initiates corrective actions, and implements protective actions for onsite personnel. In the event that the Station Emergency Director becomes incapacitated and can no longer fulfill the designated responsibilities, the Operations Manager will assume the responsibilities of the Station Emergency Director until relieved by another qualified Station Emergency Director.

C) Correct. Per EP-AA-112-200 TSC Activation and Operation, the Station Emergency Director supervises and directs the station emergency response organization. The Station Emergency Director's responsibilities include organizing and coordinating onsite emergency efforts. Additionally, the Station Emergency Director has requisite authority, plant operating experience and qualifications to implement inplant recovery operations.

D) Incorrect. Plausible as the Corporate Emergency Director is an actual EOF ERO position, The Corporate Emergency Director is the designated individual who has the authority, management ability, and technical knowledge to manage Exelon Nuclear's Emergency Response activities in the Emergency Operations Facility (EOF). The EOF shall achieve Minimum Staffing and facility activation within 60 minutes of an Alert or higher declaration.

Technical Reference(s): EP-AA-112-200, “TSC Activation and Operation,” Rev

Proposed references to be provided to applicants during examination: N/A

Learning Objective:

Question Source:	Bank #	X	
	Modified Bank #	___	(Note changes or attach parent)
	New	___	
Question History:	Last NRC Exam	Clinton 2019 NRC ILE (Q73)	
Question Cognitive Level:	Memory or Fundamental Knowledge	2	
	Comprehension or Analysis	___	
10 CFR Part 55 Content:	41.10 / 45.13		

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.4.12	
	Importance Rating	4.0	

K/A Statement: Knowledge of general operating crew responsibilities during emergency operations.

Proposed Question: **Question #75**

During an emergency condition, Reactor Operator actions that DEVIATE from plant Technical Specifications are needed to protect the health and safety of the public.

In accordance with HU-AA-104-101, “Procedure Use and Adherence”, these actions require…

A. approval of the Plant Manager (non-licensed).

B. approval of a licensed Senior Reactor Operator.

C. concurrence of a second licensed Reactor Operator.

D. approval of the Nuclear Regulatory Commission (NRC).

Proposed Answer: B

Explanation:

10CFR50.54:
(x) A licensee may take reasonable action that departs from a license condition or a technical specification (contained in a license issued under this part) in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with license conditions and technical specifications that can provide adequate or equivalent protection is immediately apparent.

(y) Licensee action permitted by paragraph (x) of this section shall be approved, as a minimum, by a licensed senior operator, or, at a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted, by either a licensed senior operator or a certified fuel handler, prior to taking the action.

A) Incorrect. Plausible because the Plant Manager is responsible for safe and reliable plant operation, but not required to be notified prior to taking the actions.
B) Correct. The licensee may take reasonable action that departs from a license condition or a Technical Specification in an emergency when:
1. The action is immediately needed to protect the public health and safety, and
2. No action consistent with license conditions and Technical Specifications that can provide adequate or equivalent protection is immediately apparent, and
3. As a minimum a licensed Senior Reactor Operator has approved the licensee action prior to taking the action.
C) Incorrect. Plausible because during non-transient conditions, any knowledge-based decision executed by Operations shall be peer checked.
D) Incorrect. Plausible because NRC notification is required, but the NRC is not required to be notified prior to taking the actions.

Technical Reference(s): HU-AA-104-101 Rev. 7 (p. 10)

Proposed references to be provided to applicants during examination: N/A

Learning Objective: 638.00.10

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

Question History: Last NRC Exam LaSalle 2018 ILT Exam (Q74)

Question Cognitive Level: Memory or Fundamental Knowledge 2
Comprehension or Analysis ___

10 CFR Part 55 Content: 55.41 **10**
55.43 ____

Comments: