

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

1

ID: 2544366

Points: 1.00

Unit 1 is at 100% RTP with all OPRM's INOPERABLE.

- 1A reactor recirculation pump trips
- APRM's are reading 55% with noise levels 2.5 times normal
- The LPRM noise levels are 3 times normal
- Plant conditions have plotted in region 1 on the Power-To-Flow map

What is the FIRST required action?

- A. Insert a manual scram.
- B. Implement Single Loop Thermal limits ONLY.
- C. Insert cram array control rods, and fully open FCV on running Recirc pump.
- D. Implement Single Loop Thermal limits and insert control rods to exit region 1.

Answer: A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Insert a manual scram of the reactor." is the correct answer.

With OPRM(s) trip capability "NOT MAINTAINED" Actions require checking core flow outside of region 1 of Attachment "A", if not outside region 1, then scram the reactor and exit to LGP 3-2

Distractor 1 is plausible because implementing single loop thermal limits is a required TS action during MODE 1 and 2, but incorrect because conditions plot in region 1 and OPRMs are inoperable requiring a reactor Scram.

Distractor 2 is plausible because inserting Cram arrays and raising flow are actions that would be taken to maintain the OPRM high alarm clear, but incorrect because OPRM trip capability is not maintained with OPRMs INOPERABLE.

Distractor 3 is plausible because inserting control rods to exit region 1 and implementing single loop thermal limits are required if OPRMs are operable and conditions plot within region 1 incorrect because OPRM trip capability is not maintained.

Reference LOA-RR-101 UNIT 1, REACTOR RECIRCULATION SYSTEM ABNORMAL, Revision 46,

Reference Provided during the examination: None

10 CFR 41(b)(5)

PRA: No

Cognitive Level High

Importance Level: (RO/SRO): 3.6

Tier: 1 Group: 1

K/A Number: 295001 AK2.10 Knowledge of the relationship between the (APE 1) PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION and the following systems or components: Power range monitoring system

Safety Function: 7 - instrumentation

Question Source: Bank

Question History:

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

SRO Justification: N/A

Comments:

Associated Objective: 206.00.06

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

2

ID: 2544503

Points: 1.00

Unit 1 is at 100% RTP.

- Annunciator 1PM01J-A214, 4KV BUS 141X/Y Overcurrent, alarms
- Bus 141X current is 0 amps
- Bus 141Y current is 200 amps
- B Circulating (CW) pump discharge is 7 psig and lowering 0.1 psig/min
- Condenser backpressure is 6.3" Hg and degrading 0.1" Hg/min

(1) What is the required operator action?

(2) Why?

- A. (1) Lower reactor power
(2) To mitigate the loss of condenser vacuum
- B. (1) Lower reactor power
(2) To prevent CW pump operating at runout
- C. (1) Lower CW discharge pressure to 6.5 psig by throttling CW gates
(2) To mitigate the loss of condenser vacuum
- D. (1) Lower CW discharge pressure to 6.5 psig by throttling CW gates
(2) To prevent CW pump operating at runout

Answer: A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer explanation: "(1) Lower reactor power (2) To mitigate the loss of condenser vacuum" is the correct answer.

Per LOA-AP-101 a power reduction is required as necessary to maintain condenser vacuum. Per LOA-CW-101 Attachment A with the lowest allowed operating temperature of 45°F a reduction to one running CW pump will require load reduction from 100% RTP to maintain condenser vacuum. A loss of power to 141X will cause a loss of power to 2 of 3 CW pumps and will require a power reduction to maintain condenser vacuum.

Distractor 1 is plausible because lowering reactor power is the correct action and will reduce the entire steam plant throughput, but incorrect because runout conditions will occur when CW pump discharge pressure drops to or below 6.5 psig.

Distractor 2 is plausible because throttling the CW gates to ensure maximum output from the CW pump will aid in maintaining condenser vacuum, but incorrect because CW gates are throttled to maintain discharge pressure greater than 6.5 psig.

Distractor 3 is plausible because the CW gates are throttled to prevent CW pumps from operating at runout conditions, but incorrect because the gates will be throttled to maintain discharge pressure greater than 6.5 psig.

Reference: OP-LA-103-102-1002 rev 26, LOA-AP-101 rev 64, LOP-AP-141X rev 16, LOA-CW-101 rev 29

Reference provided during examination: None

Cognitive level: High
10CFR Part 55: 41.5

Importance Level (RO/SRO): 4.2
Tier: 1 Group: 1

PRA: Yes

K/A: G2.2.44 - (APE 3) PARTIAL OR COMPLETE LOSS OF AC POWER -- EQUIPMENT CONTROL Ability to interpret control room indications to verify the status and operation of a system and understand how operator actions and directives affect plant and system conditions

Safety Function: 6 Electrical

SRO Justification: N/A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Question Source: New

Question History:

Comments:

Objective #: 111.00.13

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

3

ID: 2508362

Points: 1.00

Unit 2 is at 100% RTP when a loss of DC Bus 213 occurs.

Before any action is taken a LOCA occurs on Unit 2.

- DW pressure is 6 psig and rising
- RPV water level is 30" and stable

The Bus 213 supply breaker was inadvertently bumped to OFF and is ready to be re-closed.

What action is required prior to closing the Bus 213 supply breaker and resetting HPCS logic, and what is the reason for that action?

- A. Place 2B DG in Maintenance; To prevent HPCS trip on 2B DG load shed
- B. Place HPCS pump in Pull-to-Lock; To prevent HPCS auto-start
- C. Place HPCS pump in Normal-after-Start; To prevent HPCS pump trip
- D. Place HPCS pump in Normal-after-Trip; To prevent HPCS auto-start

Answer: B

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "(1) Place HPCS pump in Pull-to-Lock (2) To prevent HPCS auto-start" is the correct answer.

Per LOA-DC-201, When Bus 213 is de-energized the initiation logic for HPCS is disabled, Bus 213 undervoltage relay is actuated. The LOCA signals (low reactor level and Hi DW pressure) will not auto start the HPCS pump. HPCS will auto start and inject when the breaker is closed and logic is reset as the logic when reset will detect Hi DW pressure signaling an auto start and inject signal for HPCS. Per LOA-DC-201, Sec B.8 the HPCS pump should be placed in pull to lock prior to re-energizing Bus 213 and resetting the logic.

Distractor 1 is plausible because Drywell pressure is above 1.66 psig and restoring power will cause the 2B DG to auto-start, but incorrect because with Bus 213 de-energized HPCS will not start.

Distractor 2 is plausible because the HPCS control switch is a 4 position switch with Normal-after-Start as a unique position that picks up control and indication relays, but incorrect because with Bus 213 de-energized HPCS will not start.

Distractor 3 is plausible because with a sealed in initiation signal ECCS pump auto start can be overridden by placing the control switch to trip and allowing it to return to Normal-after-Trip, but incorrect because resetting the HPCS logic after restoring power to Bus 213 will cause the HPCS pump to auto-start on high Drywell pressure.

Reference: LOA-DC-201 Rev 24; 1E-2-4222AB; 1E-2-4223AB; 1E-2-4223AC, 1E-2-4223AE

Reference provided during examination: None

Cognitive level: High

10CFR Part 55: 41.5 / 41.7

Importance Level (RO/SRO): 3.7

Tier: 1 Group: 1

PRA: Yes

K/A: 295004 AK1.06 - Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the (APE 4) PARTIAL OR COMPLETE LOSS OF DC POWER: Prevention of inadvertent system(s) actuation upon restoration of DC power

Safety Function: 6 - Electrical

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

SRO Justification: N/A

Question Source: New For ILT 23-1 NRC exam

Question History: New

Comments:

Objective #: 006.00.14

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

4

ID: 2544506

Points: 1.00

Unit 1 is at 100% RTP.

A failure of the 1A TDRFP Manual/Auto (M/A) Speed Control System results in a maximum demand to the pump.

- 1A TDRFP is placed in manual
- RPV water level is 60 inches and steady
- Unit 1 remains at 100% RTP

What is the NEXT required operator action?

- A. Trip 1A TDRFP ONLY.
- B. Scram and trip Main Turbine and both TDRFPs.
- C. Lower RPV water level to +20 to +50 inches ONLY.
- D. Scram and close Main Steam Isolation Valves (MSIVs).

Answer: B

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Scram and trip Main Turbine and both TDRFPs." is the correct answer.

The high reactor water level trips in both Chan B and C will result in a trip of the feed pumps and the main turbine. With power greater than 25% the reactor should automatically scram on the turbine trip.

Per LOR-1H13-P603-A310 "FW CONTROL RX VESSEL LVL 8 TRIP" the high level trip setpoint is 59.5" TS (or 55" NFS) and current RPV level is 60 inches with the TDRFPs and the Main turbine online. The required operator action is to perform the automatic action that failed to occur. Since reactor power is greater than 25% a Scram is also required per LOA-TG-101.

Distractor 1 is plausible because malfunctioning equipment is removed from service if not needed for continued operation, but incorrect because all TDRFPs need to be tripped.

Distractor 2 is plausible because RPV water level is outside the normal control band of 20-50 inches with feedwater in control requiring water level to be lowered back to the normal control band per LOA-FW-101, but incorrect because a Scram and trip of turbine driven components is also required due to the trip condition.

Distractor 3 is plausible because MSIVs will be closed on a high RPV water level with direction given from the same LOR-1H13-P603-A310, but incorrect because this action is not required until RPV water level is greater than 106 inches on the upset range.

Reference: LOR-1H13-P603-A310 FW CONTROL RX VESSEL LVL 8 TRIP rev 7, LOA-TG-101 rev 25, LOA-FW-101 rev 13

Reference provided during examination: None.

Cognitive level: High
10CFR Part 55: 41.(b)(10)

Importance Level (RO/SRO): 4.4
Tier: 1 Group: 1

PRA: No

K/A: 295005 AA2.07 - Ability to determine or interpret the following as they apply to (APE 5) MAIN TURBINE GENERATOR TRIP: Reactor water level
Safety Function: 4
SRO Justification: N/A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Question Source: New

Question History: N/A

Comments:

Objective #:031.00.12

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

5

ID: 2545889

Points: 1.00

Unit 1 is at 100% RTP.

A manual scram is inserted.

All control rods are inserted.

5 minutes after the scram how will operators control reactor pressure?

- A. Safety Relief Valves
- B. Main Turbine Bypass Valves ONLY
- C. Main Turbine Control Valves ONLY
- D. Main Turbine Bypass Valves and Control Valves

Answer: B

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Main Turbine Bypass Valves ONLY" is the correct answer.

Per LGA-001 "RPV Control" the first action following meeting an entry condition, in this case an RPV water level below 11 in. will occur following a manual scram from 100% RTP, is to scram and enter LGP-3-2.

LGP-3-2 directs performance of the Attachment E-1 hardcard. The direction to stabilize RPV pressure below 1020 psig with the main condenser available is to use the bypass valves.

Distractor 1 is plausible because safety relief valves are used for RPV pressure control, but incorrect because SRVs are classified as an alternate pressure control system, figure P1, in the LGAs and the main condenser and bypass valves are available.

Distractor 2 is plausible because turbine control valves are the normal online method to control RPV pressure, but incorrect because the Main Turbine will trip on reverse power following a scram and the TCVs will close on the turbine trip.

Distractor 3 is plausible because Bypass valves will be used to control RPV pressure, but incorrect because the Main Turbine will trip on reverse power following a scram and the TCVs will close on the turbine trip.

Reference: LGA-001 rev 19; LGP-3-2 rev 81
References provided during examination: None

Cognitive level: Memory
10CFR part 55: 41(b)(5)

Importance level (RO/SRO): 4.1
Tier: 1 Group: 1

PRA: No

K/A: K/A: 295006 AK1.04 - Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the (APE 6) SCRAM: Pressure control

Safety Function: 3

SRO Justification: N/A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Question Source: Bank

Question History: N/A

Comments:

Objective: 410.00.01

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

6

ID: 2508600

Points: 1.00

Unit 1 and 2 are at 100% RTP.

Evacuation of essential personnel from the Main Control Room is required when...

- A. all fans trip on the operating VC/VE train and cannot be restarted.
- B. 15 ppm ammonia is detected in the control room ventilation system.
- C. a probable airborne terrorist threat has been confirmed by the NRC.
- D. there is a fire at 1H13-P603 that cannot be extinguished with a fire extinguisher.

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "there is a fire at 1H13-P603 that cannot be extinguished with a fire extinguisher." is the correct answer.

The time to evacuate the control room will be primarily based on the habitability of the control room followed by the ability to control critical plant equipment necessary to shutdown the reactor and initiate and maintain long term core cooling. Some Control Room Atmospheric conditions, such as high ammonia concentration which may be above IDLH, may require SCBA use while the control room atmosphere is restored. However, in this situation the source of the IDLH atmosphere is the fire in the control room which will not be corrected until the fire is extinguished.

A fire that cannot be extinguished using the entire contents of a portable fire extinguisher is beyond the incipient phase and into the growth phase which will require fire brigade and the use of higher capacity fire extinguishing systems.

Per LOA-FX-101 with a fire in the MCR with or without a loss of offsite power, the control room is evacuated to ensure safe shutdown.

Per UFSAR 7.4.4

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

7.4.4. Shutdown Outside the Control Room

It is possible to shut down the reactor from outside the main control room and bring the reactor to cold conditions in an orderly fashion, in compliance with General Design Criterion 19 of 10 CFR 50, Appendix A.

7.4.4.1 Conditions Assumed to Exist as the Main Control Room Becomes Inaccessible

- a. The plant is operating initially at, or less than, design power.
- b. Loss of offsite a-c power is considered unlikely but credible. The remote shutdown panel is powered from a Class 1E power system bus so backup a-c power is automatically supplied by the diesel generator in the event of loss of offsite power. Manual controls of the diesel generator are also available locally.
- c. A loss-of-coolant accident is not assumed, so that complete control of engineered safeguards systems from outside the control room shall not be required.
- d. Plant personnel evacuate the control room.
- e. The control room continues to be inaccessible for several hours.
- f. The event that causes the control room to become inaccessible is assumed to be such that the operator can manually scram the reactor before leaving the main control room. If this is not practical, opening the output breakers of the RPS logic can be used as a backup means to achieve reactor shutdown.

Distractor 1 is plausible because a loss of ventilation to an enclosed space could lead to degrading atmospheric conditions, but incorrect because the MCR won't be evacuated until an IDLH atmosphere is reached.

Distractor 2 is plausible because high ammonia levels can create an IDLH environment, but incorrect because ammonia levels are below IDLH (<300 ppm), CR HVAC will be placed into recirculation mode on high ammonia levels (>12.5 ppm) and MCR personnel have a TCA to don respiratory equipment when ammonia is detected but they are not directed to evacuate the MCR.

Distractor 3 is plausible because following Sep 11, 2001, NRC updated guidance requiring stations to have contingencies related to damage to large areas of the plant due to explosions that could be caused by aircraft, but incorrect because the MCR will not be evacuated until absolutely necessary.

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Reference: LOA-FX-101, Rev 33; UFSAR 7.4.4; LOR-1PM06J-B204 rev 7;
LOA-RX-101 rev 13

Reference provided during examination: None

Cognitive level: Low

10CFR Part 55: 41.5 / 41.10

Importance Level (RO/SRO): 4.1

Tier: 1 Group: 1

PRA: No

K/A: 295016 AK3.04 Knowledge of the reasons for the following responses or actions as they apply to (APE 16) CONTROL ROOM ABANDONMENT:
Abandonment criteria

Safety Function: 7 Instrumentation

SRO Justification: N/A

Question Source: New

Question History: N/A

Comments:

Objective #: 054.00.15

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

7

ID: 2544507

Points: 1.00

Unit 1 is in MODE 4.

- Reactor Vessel temperature is 170°F and steady
- Reactor Water level is +45 inches controlled by condensate flow through the LFFRV
- Reactor Water Cleanup system (RT) is rejecting to radwaste IAW LOP-RT-09, RWCU System Coolant Rejection
- 1A RHR pump is in shutdown cooling

Subsequently:

- The common, 1A, and 1B Reactor Building Closed Cooling Water (WR) pumps trip and cannot be restarted.

Which load is required to be secured?

- A. 1A RHR Pump
- B. Fuel Pool Cooling Pumps
- C. Unit 1 Station Air Compressor
- D. Reactor Water Cleanup System

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Reactor water Cleanup System" is the correct answer.

The Reactor Building Closed Cooling Water System (RBCCW) provides cooling for the RT System non-regenerative heat exchangers and RT Pump Motor heat exchangers (one/pump). RBCCW System temperature increases could affect the RT System in that the outlet of the non-regenerative heat exchanger could increase to the high temperature alarm associated with the inlet to the filter/demineralizer and possibly cause the RT System isolation valve to close at 140°F.

Per LOA-WR-101 LOSS OF REACTOR BUILDING CLOSED COOLING WATER (RBCCW) rev 16 a loss of RBCCW system requires shutting down RBCCW loads. Of the listed loads only the RWCU system is cooled by RBCCW

Distractor 1 is plausible because the RHR system relies upon area coolers to remove the heat generated by the RHR system operation, but incorrect because these area coolers are supplied by the Diesel Generator Cooling water system which is not impacted by the loss of RBCCW.

Distractor 2 is plausible because the station air compressors are supplied by the component cooling water system and will trip on high temperature if a loss of cooling occurs, but incorrect because the station air compressors are supplied cooling from the turbine building closed cooling water system which is not impacted by the loss of RBCCW.

Distractor 3 is plausible because the Fuel Pool Cooling pumps are secured when a loss of cooling water has been lost to restore cooling water pressure, but incorrect because the cooling water for the Fuel Pool Cooling system is the Service Water system which is not impacted by the loss of RBCCW.

Reference LOA-WR-101 rev 16; LOA-WT-101 rev 5, UFSAR 9.2, LOA-WS-101 rev 7

Reference provided during examination: None

Cognitive level: Memory
10CFR Part 55: 41.8 / 41.10

Safety Function: 8
Level (RO/SRO): 3.5
Tier: 1 Group: 1

PRA: No

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

K/A: 295018 AK2.03 Knowledge of the relationship between the (APE 18)
PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW)
and the following systems or components: Reactor water cleanup system

SRO Justification: N/A

Question Source: New

Question History:

Comments:

Objective # 114.029

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

8

ID: 2504970

Points: 1.00

Unit 1 is at 100% RTP.

The Instrument Air line to the on-line CRD Flow Control Valve, 1C11-F002A, becomes sheared at the valve actuator.

(1) What is the required action?

(2) CRD cooling water flow is at...

- A. (1) Swap to CRD flow control valve B, 1C11-F002B
(2) MINIMUM
- B. (1) Swap to CRD flow control valve B, 1C11-F002B
(2) MAXIMUM
- C. (1) Control flow with 1C11-R600, CRD SYS Flow controller, in manual
(2) MINIMUM
- D. (1) Control flow with 1C11-R600, CRD SYS Flow controller, in manual
(2) MAXIMUM

Answer: A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "(1) Swap to CRD flow control valve B, 1C11-F002B (2) MINIMUM" is the correct answer.

Due to the loss of Instrument Air (IA) the on-line flow control valve fails closed. The valve will not fully close as there is a mechanical stop limiting the valve closure to prevent drive and cooling water pressure issues.

Per LOP-RD-06 CONTROL ROD DRIVE SYSTEM FLOW CONTROL VALVE TRANSFER

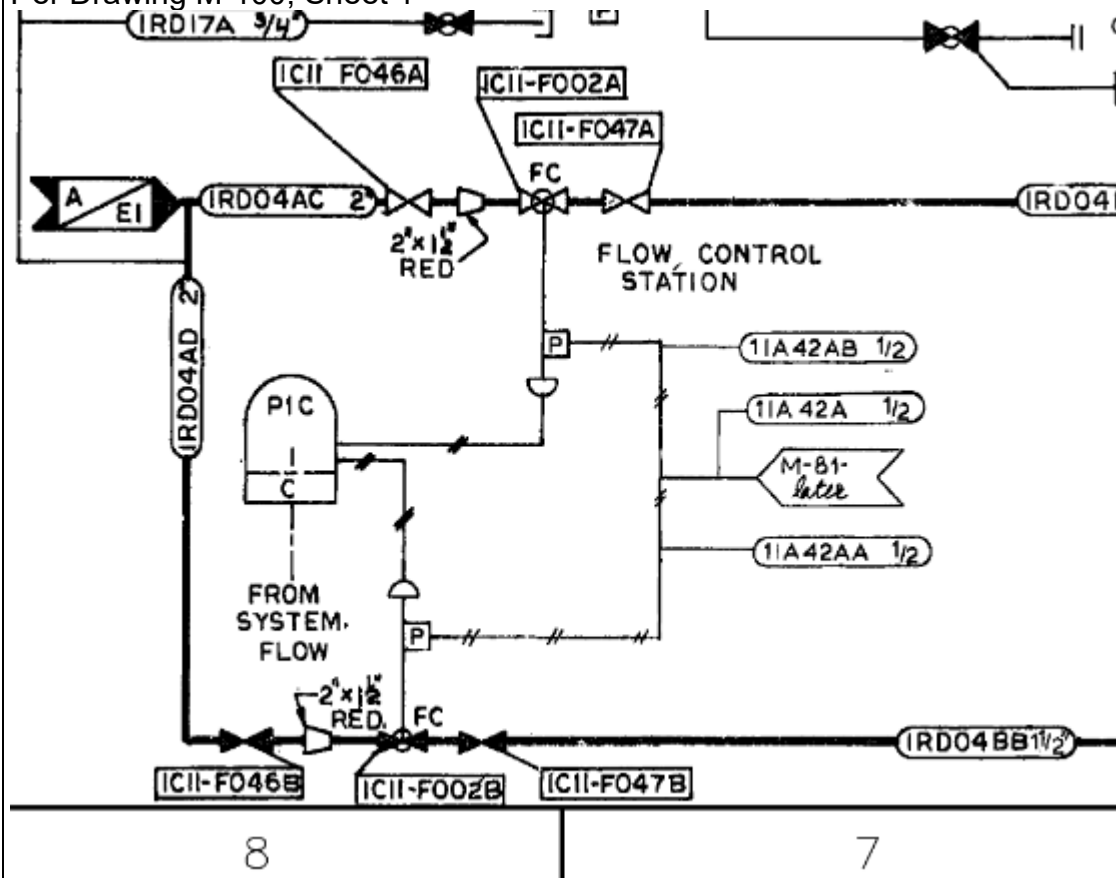
D.5 All expected flow conditions may not be met during valve transfer when shutdown with low flow. If valve is at its minimum open mechanical stop position, then consider the condition met.

Level of Use
Continuous

4 of 14

LOP-RD-06
Revision 15
July 16, 2013

Per Drawing M-100, Sheet 1



Distractor 1 is plausible because swapping to the B CRD flow control valve and too high of cooling water flow can lead to high cooling water differential pressure which cause rods to drift (LOP-RD-08 D.8), but incorrect because instrument air

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

is used to open the CRD flow control valve and a loss of air will cause the CRD flow control valve to drift to a mechanical stop.

Distractor 2 is plausible because CRD flow will lower to minimum and some systems use instrument air to provide indication and 1C11-N004, CRD flow transmitter, inputs to 1C11-R600 for controlling constant flow in automatic which would require 1C11-R600 in manual to output a signal to maintain constant position of 1C11-F002A on a loss of flow signal, but incorrect because the flow signal is generated by a venturi type flow element, 1C11-N003.

Distractor 3 is plausible because some systems use instrument air to provide indication and 1C11-N004, CRD flow transmitter, inputs to 1C11-R600 for controlling constant flow in automatic which would require 1C11-R600 in manual to output a signal to maintain constant position of 1C11-F002A on loss of flow signal and some air operated valves fail full open on loss of air, but incorrect because CRD FCV is an air opened valve that will fail closed to a mechanical stop on a loss of IA.

Reference: LOA-RD-101 rev 26; LOP-RD-06 CONTROL ROD DRIVE SYSTEM FLOW CONTROL VALVE TRANSFER rev 15; Dwg M-100, Sht 1; LOP-RD-08 rev 32; M-2100 sheet 1 rev L;

Reference provided during examination: None

Safety Function: 1

Cognitive level: High

10CFR Part 55: 41.(b)(7)

Level (RO/SRO): 4.0

Tier: 1 Group: 1

PRA: No

K/A: 295019 AK2.01: Knowledge of the relationship between Partial or Complete Loss of Instrument Air and the following systems or components:
CRD hydraulics

SRO Justification: N/A

Question Source: New

Question History:

Comments:

Objective #025.00.13

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

9

ID: 2508749

Points: 1.00

Unit 1 is in MODE 3.

- A RHR is in Shutdown Cooling (SDC)
- B RR pump is running
- RWCU is in operation
- RCS temperature is 241°F and lowering 0.1°F/min
- RHR service water flow is 7400 gpm

The following occurs:

- 1A RHR WS pump trips and emits a strong acrid smell
- RHR service water flow is 5800 gpm and stable
- RCS temperature is 241°F and rising 0.1°F/min

Which action will re-establish an RPV cooldown?

- A. Start the B RHR pump ONLY
- B. Start the standby Reactor water cleanup (RWCU) pump
- C. Transfer shutdown cooling operation to the B RHR train
- D. Throttle the 1B RHR SW pump discharge valve ONLY

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Transfer shutdown cooling operation to the B RHR Heat Exchanger" is the correct answer.

RHR service water pumps are rated to 4000 gpm. With RHR WS flow at 5800 gpm the service water pump is operating at 45% above its design flowrate and is at or near runout conditions. With the loss of one service water pump in the A loop of RHR, the RHR service water flow at 5800 GPM and a heatup in progress, in order to re-establish a cooldown Shutdown Cooling needs to be transferred to the B RHR service water loop.

Per LOA-RH-101

- | | | | |
|-----|--|------|---|
| 15. | CHECK proper operation/lineup of RHR Service Water loop. | 15.1 | <u>If</u> all Traveling Screens are blocked, UNLOCK and OPEN 0E12-F300, CSCS Equipment Cooling Bypass Valve. |
| | | 15.2 | RESTORE RHR Service Water per LOP-RH-05, Operation of the RHR Service Water System. |
| | | 15.3 | <u>IF</u> the operating RHR Service Water loop is inoperable, CONSIDER SHIFTING to the standby RHR Shutdown Cooling and RHR SW loops per LOP-RH-07, Shutdown Cooling System Startup, Operation, and Transfer. |

Distractor 1 is plausible because SDC must be transferred to the B loop of RHR, but incorrect because more steps are involved than starting the B RHR pump.

Distractor 2 is plausible because placing a second component on-line in the Reactor Water Cleanup System is the method to establish alternate decay heat removal with reactor water cleanup as directed out of LOA-RH-101 and starting a second pump in parallel will raise system flow rate, but incorrect because RWCU flow is maintained by the demineralizer outlet valves in automatic.

Distractor 3 is plausible because use of a single RHR SW pump operation is an acceptable operational lineup for RHR service water requiring throttling the RHR WS discharge valve, but incorrect because the RHR SW flow required to establish a cooldown rate is beyond the operational capability of a single RHR service water pump and the RHR WS discharge valve is normally full open and is throttled closed to prevent pump runout.

Reference: LOA-RH-101, Rev 24; Test Curve 17053 rev A, performance curve 19515 rev 03

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Reference provided during examination: None

Cognitive level: High

10CFR Part 55: 41.5 / 41.7

Importance Level (RO/SRO): 3.4

Tier: 1 Group: 1

PRA: No

K/A: 295021 AA1.03 Ability to operate or monitor the following as they apply to (APE 21) LOSS OF SHUTDOWN COOLING: Component cooling water systems

Safety Function: 4

SRO Justification: N/A

Question Source: New

Question History:

Comments:

Objective #: 064.00.13

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

10

ID: 2545209

Points: 1.00

Unit 2 is in Mode 5 and refueling the reactor is in progress.

- An irradiated fuel bundle is dropped on top of the reactor core
- PCIS group 4 initiated on Fuel Pool (FC) exhaust radiation levels
- Unit 2 Standby Gas Treatment (VG) starts
- Unit 1 Standby Gas Treatment started and was secured
- Unit 2 Reactor Building Ventilation (VR) continues operating
- Unit 1 Reactor Building Ventilation is isolated
- Unit 2 Reactor Building (RB) DP is -0.5" H₂O

Which action is required and why?

- A. Secure VG as RB DP is too negative for VG operation
- B. Secure VR as RB DP is too negative for VG operation
- C. Secure VG to ensure plant effluent is filtered prior to release
- D. Secure VR to ensure plant effluent is filtered prior to release

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Secure VR to ensure plant effluent is processed prior to release" is the correct answer.

The damaged fuel assemblies with release radioactive gases to the refuel floor area. The process radiation monitors in the FC ventilation exhaust line will actuate a Group 4 isolation at 10 mRem/hr which will isolate Reactor Building ventilation and start Standby Gas Treatment (SBGT).

Per LGA-002 if RB ventilation isolated or should have isolated then RB vent isolations are verified, and VG operation is verified with LGA-VG-201.

Per LGA-VG-201 both trains of VG and VQ are used to maintain RB DP.

Distractor 1 is plausible because VG is designed to maintain RB DP at -0.25" H₂O and if DP is too high the unit without the high radiation ventilation trip will secure VG, but incorrect because VR should have isolated and the RB DP will be assessed following shutdown of VR which is assisting maintaining the RB DP negative.

Distractor 2 is plausible because removing VR from service will raise RB DP, but incorrect because this action is required because VR should have isolated to send all the ventilation exhaust through VG for processing prior to release.

Distractor 3 is plausible because VR is operated in LGA-002 through overrides when VG and VQ cannot maintain RB DP below 0" H₂O to prevent unmonitored releases from the RB, but incorrect because VR should have isolated on the PCIS group 4 isolation and will not be started to control RB DP until VG AND VQ are in operation with RB DP greater than 0" H₂O.

Reference: UFSAR 15.7.4; LOR-2H13-P601-F205, Rev 6; LGA-002-5 rev 0; LGA-VG-201 rev 5

Reference provided during examination: None

Cognitive level: High

10CFR Part 55: 41.5

Importance Level (RO/SRO): 3.7

Tier: 1 Group: 1

PRA: No

K/A: 295023 AK3.03 - Knowledge of the reasons for the following responses or actions as they apply to (APE 23) REFUELING ACCIDENTS: Ventilation isolation

Safety Function: 9 - Rad Release

SRO Justification: N/A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Question Source: New

Question History: N/A

Comments:

Objective #: 52.00.10

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

11

ID: 2508863

Points: 1.00

Unit 1 is at 100% RTP when the 1A Primary Containment Chiller trips.

Actions to restore 1A Primary Containment Chill Water are in progress.

- DW pressure rose to 2.0 psig then lowered to 1.6 psig
- RPS channel A de-energized
- RPS channel B remains energized
- HPCS started automatically
- Common, 1A, and 1B DGs started automatically

What is the NEXT required operator action?

- A. Secure HPCS
- B. Secure the 0 and 1A Diesel Generators
- C. Insert a reactor scram for RPS channel B
- D. Reset the reactor scram on RPS Channel A

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Insert a reactor scram for RPS channel B" is the correct answer.

Per LOR-1H13-P603-B405 The RPS trip setpoint for HI DW Pressure is 1.8 psig (1.93 psig TS allowable). Since DW pressure has exceeded the RPS setpoint there is an indication the B channel of RPS has failed to trip. The required action is to place the trip system in the required condition, TRIPPED.

Distractor 1 is plausible because the drywell pressure is currently below the HPCS initiation pressure and HPCS is secured if initiated at 100% RTP, but incorrect because the failure of B RPS to scram on high drywell pressure will be addressed first.

Distractor 2 is plausible because B RPS did not trip and the drywell pressure is now below the RPS scram setpoint, but incorrect because drywell pressure exceeded the RPS scram setpoint and B RPS has failed to trip.

Distractor 3 is plausible because DW pressure is below the ECCS initiation setpoint and equipment not needed to combat the transient can be secured, but incorrect because B RPS failing to trip will be addressed first.

Reference: LOR-1H13-P603-B405, Rev 3; LOR-1H13-P601-A106, Rev 3; LOR-1H13-P603-B501, Rev 4

Reference provided during examination: None

Cognitive level: High
10CFR Part 55: 41.7

Importance Level (RO/SRO): 4.2
Tier: 1 Group: 1

PRA: No
K/A: 295024 EA1.05 - Ability to operate or monitor the following as they apply to (EPE 1) HIGH DRYWELL PRESSURE: RPS

Safety Function: 6 - instrumentation
SRO Justification: N/A

Question Source: New
Question History:

Comments:

Objective #: 049.00.11

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

12

ID: 2508892

Points: 1.00

Unit 1 is operating at 100% RTP.

- Reactor pressure is 1010 psig and rising at 10 psig/min
- EHC remains in Dome Pressure Control mode at 923 psig

What is the next REQUIRED operator action?

- A. Trip EHC pumps
- B. Manually scram reactor
- C. Swap EHC to standby Throttle Pressure Control mode
- D. Commence load drop to maintain pressure within normal operating limits

Answer: B

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Manually Scram reactor" is the correct answer. When EHC is not controlling reactor pressure the required immediate action is to insert a manual scram per LOA-EH-101.

Per LOA-EH-101



MANUAL SCRAMS



- o Reactor pressure can NOT be controlled.
- o More than one TCV/TSV stuck closed.

B. <u>ACTION/EXPECTED RESPONSE</u>		<u>RESPONSE NOT OBTAINED</u>	
B.1 <u>Loss of Pressure Regulator (Erratic TCV motion or Pressure Regulator Failure)</u>			
Immediate Action	1. CHECK EHC is CONTROLLING reactor pressure.	1.1 MANUALLY SCRAM reactor.	1.2 EXIT to LGP-3-2.
	2. CHECK reactor pressure BELOW 1020 PSIG.	2.1	REDUCE reactor pressure to below 1020 psig within 15 minutes and refer to Tech Spec 3.4.12.
	3. CHECK reactor pressure - WITHIN the "Allowed Region" shown on ATTACHMENT A "Reactor Pressure vs. Core Thermal Power."	3.1	ENTER Tech Specs 3.2.1, 3.2.2 and/or 3.2.3 Condition A.
		3.2	ADJUST EHC Active Pressure setpoint (Throttle or Dome) to maintain reactor pressure in the "Allowed Region" shown on ATTACHMENT A "Reactor Pressure vs. Core Thermal Power."
	4. CHECK Reactor Pressure NOT erratic and is consistent with EHC Active Pressure Setpoint (Throttle or Dome).	4.1	Perform LOP-EH-04, Section E.5 and swap to standby pressure control mode (Dome to Throttle, or Throttle to Dome) to stabilize Reactor Pressure.

Distractor 1 is plausible because tripping EHC pumps should result in automatic reactor scram and turbine trip, but incorrect because the procedural requirement is to insert a manual scram.

Distractor 2 is plausible because swap to standby pressure control is an action in LOA-EH-101, but incorrect because the procedural requirement is to insert a manual scram when pressure is not in control.

Distractor 3 is plausible because LOA-PWR-101 directs action to lower pressure by adjusting either pressure set or lowering power if pressure is greater than or

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

equal to 1005 psig, but incorrect because the procedural requirement is to insert a manual scram when pressure is not in control.

Reference: LOA-EH-101, Rev 38

Reference provided during examination: None

Cognitive level: Memory

10CFR Part 55: 41.10

Importance Level (RO/SRO): 4.7 / 4.3

Tier: 1 Group: 1

PRA: No

K/A: 295025 EA2.01 - Ability to determine or interpret the following as they apply to (EPE 2) HIGH REACTOR PRESSURE: Reactor pressure

Safety Function: 3 - Reactor Pressure Control

SRO Justification: N/A

Question Source: New

Question History: N/A

Comments:

Objective #: 074.00.11

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

13

ID: 2508903

Points: 1.00

Unit 1 is at 100% RTP.

- 1B21-F013C SRV has stuck open
- Suppression Pool Temperature is 105.2°F and rising 0.2°F/min
- Drywell pressure is 0.6 psig and rising 0.1 psig/min
- Suppression Chamber Pressure is 0.25 psig and rising 0.1 psig/min

At this time, operators are required to _____.

- A. scram the reactor ONLY
- B. start suppression chamber sprays
- C. scram the reactor and depressurize the RPV
- D. place suppression pool cooling into operation

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: : "place two loops of Suppression pool cooling into operation" is correct.

Required actions per LGA-003.

Per LGA-003:

As suppression pool temperature is above 105°F, operators are directed to maximize suppression pool cooling and that will be met by placing two loops of Suppression Pool cooling into operation.

Distractors 1 is plausible because high drywell pressure requires a reactor plant scram, but it is incorrect because a scram is not required until 1.93 psig in the drywell.

Distractor 2 is plausible because suppression chamber sprays are required in the Primary Containment Pressure leg of LGA-003 which has been entered due to suppression pool temperature above 105 °F, but incorrect because suppression chamber sprays are not directed until >1.93 psig drywell pressure.

Distractors 3 is plausible because scrambling the reactor and RPV depressurization is a required action at higher suppression pool temperatures, but incorrect because suppression pool temperature is <120°F so conditions have not been met to require a RPV depressurization.

Reference: Technical Specification Bases 3.6.2.1; LGA-003, Rev 19

Reference provided during examination: None

Cognitive level: High

10CFR Part 55: 41.10 / 43.5

Level (RO/SRO): 4.1 / 4.0

Tier: 1 Group: 1

PRA: No

K/A: 295026 EA2.01 - Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool water temperature.

10 CFR55: 41.10

Safety Function: 3 Pressure Control

SRO Justification: N/A

Question Source: Bank

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Question History:

Comments:

Objective#: 503.00.04

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

14

ID: 2508937

Points: 1.00

Unit 1 is at 100% RTP.

- A small break LOCA has occurred
- Drywell pressure has risen to 0.9 psig and stabilized
- Drywell temperature is 150°F and rising

What action is required NEXT?

- A. Vent the drywell
- B. Start drywell sprays
- C. Trip Reactor Recirc Pumps
- D. Start all available drywell cooling

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

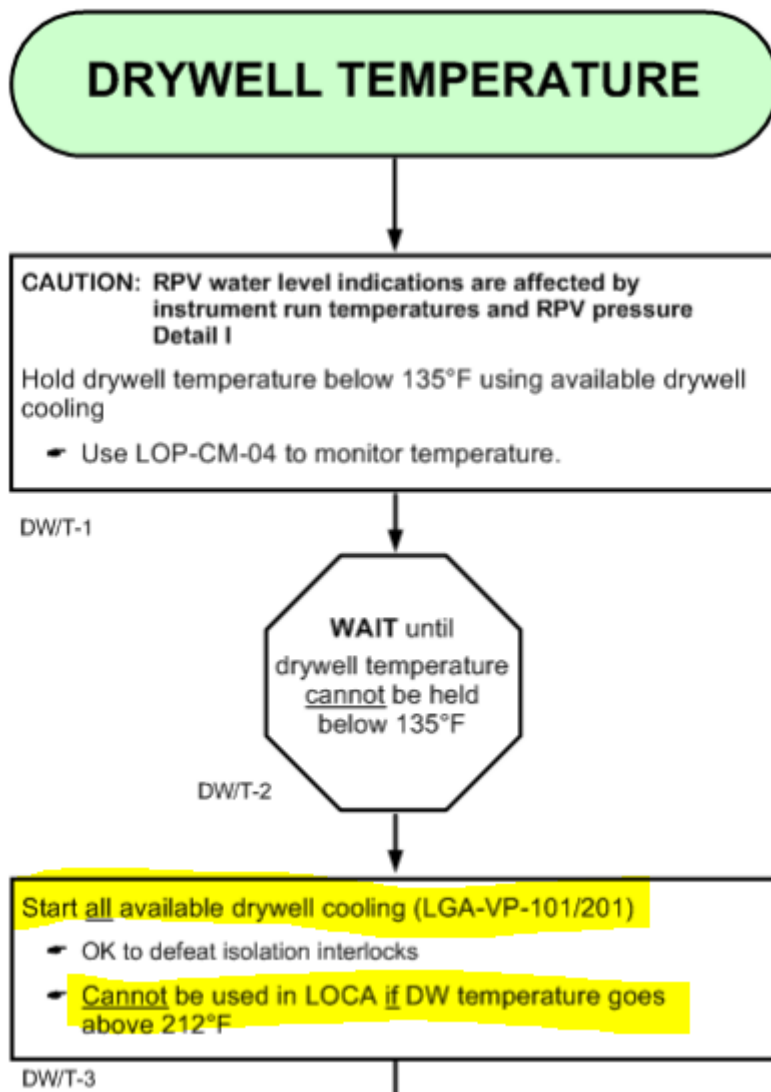
LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Start all available drywell cooling" is the correct answer.

LGA-003 states that if you cannot hold drywell temperature below 135 degrees LGA-VP-01 may be used if a LOCA has not caused drywell temperature to rise to greater than 212 degrees

Per LGA-003



Distractor 1 is plausible because this is required at elevated drywell pressures when RCIC is required for core cooling, but incorrect because DW pressure is below the level where this action may be taken.

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Distractor 2 is plausible because drywell sprays are used to lower drywell pressure later in the drywell pressure leg, but incorrect because RR Pumps must first be tripped.

Distractor 3 is plausible because RR Pumps are tripped to allow drywell sprays to be initiated, but incorrect because this is not until after all available Drywell cooling has been started and Drywell pressure is greater than the Drywell Spray initiation limit with a lowest pressure of 4.1 psig at 150°F.

Reference: LGA-003, Primary Containment Control, rev 19; LGA-VP-101, Primary Containment Temperature Reduction, rev 00
Reference provided during examination: None

Cognitive level: High
10CFR Part 55: 41.7 / 45.6

Level (RO/SRO): 3.6
Tier: 1 Group: 1

PRA: No
295028 EA1.02 - Ability to operate or monitor the following as they apply to (EPE 5) HIGH DRYWELL TEMPERATURE (MARK I AND MARK II ONLY):
Drywell ventilation system

Safety Function: 9
SRO Justification: N/A

Question Source: Bank
Question History:

Comments:

Objective # 503.00.06

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

15

ID: 2508984

Points: 1.00

In LGA-004, RPV Blowdown, why is initiating ADS prohibited when suppression pool level is below -18 feet?

- A. Chugging begins to occur in the SRV tailpipes that could result in fatigue failure of the SRV tailpipes.
- B. Opening SRVs below -18 feet will exceed allowable stresses in the SRV tailpipes and tailpipe supports.
- C. The SRV T-quenchers may be uncovered and steam suppression during an ADS blowdown can no longer be assured.
- D. Insufficient volume of water exists in the suppression pool to absorb the energy from a DBA LOCA so the maximum RPV cooldown rate cannot be exceeded.

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "The SRV T-quenchers may be uncovered and steam suppression during an ADS blowdown can no longer be assured." is the correct answer.

Per LGA-003, the lowest level that can be read in the Control Room is -18 feet and was chosen as the value at which SRVs are not used per LGA-004 because it is not assured that the SRV discharge device is covered.

Per LPGP-PSTG-02B, the top of SRV discharge T-Quencher is located at -22.6 feet suppression pool level. If the SRVs were opened with the discharge devices exposed, steam would pass directly into the suppression chamber airspace, bypassing the suppression pool. The resulting pressure increase could exceed the maximum pressure capability of the primary containment.

Distractor 1 is plausible because chugging is a phenomenon associated with the primary containment and specifically an interface between the Drywell and Suppression chamber, but incorrect because below the lowest level indication in the control room it can no longer be assured that the T-quenchers are submerged.

Distractor 2 is plausible because jet impingement and pipe-whip are associated with SRV tail pipe and Suppression pool levels and are a concern with creating a direct path to the suppression chamber following piping failure, but incorrect because uncovering the SRV T-quenchers creates a direct path to the suppression chamber airspace without requiring piping damage.

Distractor 3 is plausible because suppression pool level below -12 feet uncovers the downcomers and makes the suppression pool unable to contain all the energy release during a DBA LOCA and a cooldown rate in excess of 100°F/hr raises the probability of brittle fracture occurring, but is incorrect because when an RPV emergency depressurization is required, exceeding the 100°F/hr cooldown rate is authorized.

References: LPGP-PSTG-02C, Rev 0; LPGP-PSTG-02B Rev 0; LGA-003, Rev 19, LGA-004 rev 10

Reference provided during examination: None

Cognitive level: Memory

10 CFR Part 55 Content: 41.8 - 41.10

Level (RO/SRO): 4.0

Tier: 1 Group: 1

PRA: Yes

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

295030 EK1.05 - Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the (EPE 7) LOW SUPPRESSION POOL WATER LEVEL: SRV discharge submergence

Safety Function: 3

SRO Justification: N/A

Question Source: Bank

Question History: 17-1 ILT NRC Exam

Comments:

Objective #: 503.00.07

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

16

ID: 2504706

Points: 1.00

Unit 1 was at 100% RTP when a LOCA occurred.

- RPV pressure is 600 psig and lowering 5 psig/min
- RPV water level is -190 inches and lowering 2 in/min
- HPCS is injecting at 6000 gpm
- All SRVs are closed

What action is required to ensure adequate core cooling?

- A. Raise HPCS flow to 6200 gpm
- B. Open 1 SRV only for Minimum Core Steam Flow
- C. Stabilize RPV pressure at 550 psig using 1 BPV only
- D. Raise RPV water level to -140 inches using MDRFP

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Raise RPV water level at -160 inches using MDRFP" is the correct answer.

Per LPGP-PSTG-01C submergence is the preferred method for core cooling. The core is adequately cooled by submergence when RPV level is at or above the top of active fuel, -161 inches RPV water level.

Distractor 1 is plausible because HPCS injection must be raised from 6000 gpm to maintain ACC during DBA LOCA, but incorrect because RPV pressure must be less than or equal to 20 psig with HPCS flow greater than 6250 gpm to meet this ACC requirement.

Distractor 2 is plausible because opening SRVs to obtain MSCF is a method to ensure ACC, but incorrect because at 600 psig in the RPV 3 SRVs are required to meet MCSF.

Distractor 3 is plausible because Minimum Steam Cooling Pressure must be met to ensure ACC during steam cooling strategies and MCSF can be met using BPVs, but incorrect because 3 SRVs need to be opened to meet Minimum core steam flow at 550 psig RPV pressure and per LPGP-PSTG-01C the number of open bypass valves can be equated to an equivalent number of open SRVs which would require 3 bypass valves opened.

Reference: LGA-001 rev 19 (Figure AC); LPGP-PSTG-01C rev 0;
Reference provided during examination: none

Cognitive level: High
10CFR Part 55: 41.(b)(8)

Importance Level (RO/SRO): 4.7
Tier: 1 Group: 1

Safety Function: 2
PRA: No

K/A: 295031 Reactor Low Level EK1.01 Knowledge of the operational implications of the following concepts as they apply to Reactor Water Level:
Adequate Core Cooling

SRO Justification: N/A

Question Source: New

Question History:

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Comments:

Objective # 060.00.01

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

17

ID: 2550808

Points: 1.00

Unit 2 was at 100% RTP when an ATWS occurred.

- Reactor power is 60% RTP
- All rods remain full out

Reactor pressure control bands will be directed from...

- A. LGA-001, RPV Control
- B. LGP-3-1, Power Changes
- C. LGA-010, ATWS RPV Control
- D. LOP-EH-04, Shifting Pumps and Pressure Adjustments in the Electro Hydraulic Control System (EHC)

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "LGA-010, ATWS RPV Control" is the correct answer.

With a scram condition present and reactor power greater than 3% LGA-001 will be entered. Directions to control RPV level and prepare to maintain the main condenser as a heat sink will be directed and then LGA-001 will be exited and LGA-010 will be entered.

Power, Pressure, and Level control will then be directed from LGA-010.

Distractor 1 is plausible because LGA-001 is entered due to the ATWS condition and contains a Pressure control steps, but incorrect because on an ATWS with power greater than 3% LGA-001 is exited before any pressure control is directed from the procedure.

Distractor 2 is plausible because LGP-3-1 contains steps for lowering reactor power during normal operations and has normal pressure control bands, but incorrect because during an ATWS with power greater than 3% the emergency operating procedures will be used to direct pressure control.

Distractor 3 is plausible because the EHC system is how pressure is controlled, but incorrect because LOP-EH-04 does not contain pressure control bands for an ATWS condition.

Reference: LGA-001, Rev 19; LGA-010, Rev 20; LGP-3-1 Rev 76; LOP-EH-04 Rev 21

Reference provided during examination: None

Cognitive level: Low
10CFR Part 55: 41.10

Importance Level (RO/SRO): 3.7 / 4.3
Tier: 1 Group: 1

PRA: Yes

K/A: 295037 G2.4.5 (EPE 14) SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN // EMERGENCY PROCEDURES / PLAN Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions

Safety Function: 7 - Instrumentation
SRO Justification: N/A

Question Source: New

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Question History: N/A

Comments:

Objective #: 400.00.14

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

18

ID: 2509442

Points: 1.00

Unit 2 was at 100% RTP when a LOCA occurred.

- Primary containment Oxygen is 6% and steady
- Primary containment hydrogen is 2.5% and rising 0.1%/min
- RPV water level is 20" and lowering 1 in/min
- Primary containment purging is in progress IAW LGA-VQ-103, Unit 1 Primary Containment Purge.

Primary containment purging must be secured when...

- A. RPV water level lowers to -170 inches
- B. Primary containment hydrogen rises to 8%
- C. Offsite release rates rise to $8.27 \text{ E}+09 \text{ } \mu\text{Ci/sec}$
- D. SBTG Wide Range Gas monitor indicates $6.2\text{E}+04 \text{ } \mu\text{Ci/sec}$

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer explanation: "Offsite release rates rise to 8.27 E+09" is the correct answer.

Per LGA-VQ-103, in conditions where there is a large threat to primary containment, oxygen levels greater than 5%, plus hydrogen detected, 2%, purging is allowed as long as release rates remain below the GSEP General Emergency Level, 8.26E+09 $\mu\text{Ci/sec}$.

Distractor 1 is plausible because RPV water level in relation to the top of active fuel is a decision point in securing primary containment purge, but incorrect because if RPV water level is below the top of active fuel, -161 inches, with primary containment oxygen levels above 5% no offsite rad release limit is imposed.

Distractor 2 is plausible because hydrogen concentrations of 8% and higher are at an explosive level, but incorrect because primary containment purging is to reduce primary containment hydrogen levels.

Distractor 3 is plausible because SBGT WRGM indication of 6.10E+04 $\mu\text{Ci/sec}$ is the criteria to stop primary containment purge with oxygen levels below 5%, but incorrect because primary containment oxygen is 6%.

Reference: LGA-VQ-103 rev 4

Reference provided during examination: None

Cognitive level: Low

10CFR Part 55: 41.8 / 41.10

Importance Level (RO/SRO): 3.0

Tier: 1 Group: 1

PRA: No

K/A: 295038 EK2.09 - Knowledge of the relationship between the (EPE 15) HIGH OFFSITE RADIOACTIVITY RELEASE RATE and the following systems or components: Post accident sample system

Safety Function: 9 - radioactive release

SRO Justification: N/A

Question Source: New

Question History:

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Comments:

Objective #: 93.00.15

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

19

ID: 2545207

Points: 1.00

Unit 1 is at 15% RTP when a rupture at 1TO-H19, the Hydrogen Seal Oil Pressure Regulating Valve, occurs.

- Unit 1 Main Generator is synchronized to the Grid
- The Hydrogen Seal Oil leak is feeding a fire
- Fire header pressure is 130 psig and steady
- 0A Diesel fire pump tripped
- 0B Diesel fire pump is running

(1) What is the required action?

(2) Why?

- A. (1) Scram
(2) Emergency venting of the Main Generator is required
- B. (1) Scram
(2) Service water components need to be removed to raise fire header pressure
- C. (1) Trip the main turbine ONLY
(2) Emergency venting of the Main Generator is required
- D. (1) Trip the main turbine ONLY
(2) Service water components need to be removed to raise fire header pressure

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Trip the main turbine only; Emergency venting of the Main Generator is required" is the correct answer

Per LOA-FP-101, if a Turbine Building area fire cannot be controlled without securing Hydrogen Seal Oil, then the main turbine is removed from service in preparations for emergency venting the main generator and securing main turbine lube oil. If the main turbine Stop valve and control valve Reactor Scram is bypassed, reactor power <25%, then the Main turbine is tripped without scrambling the reactor.

Distractor 1 is plausible because LOA-FP-101 directs a Reactor shutdown/scram with a fire in progress at or above 710' elevation with Diesel Fire Pump issues and the Hydrogen Seal Oil unit is at 731' elevation in order to redirect WS for firefighting efforts, but incorrect because these actions are not required unless the fire header pressure cannot be maintained above 125 psig and given fire header pressure is at 130 psig.

Distractor 2 is plausible because LOA-FP-101 directs a Reactor shutdown/scram with a fire in progress at or above 710' elevation with Diesel Fire Pump issues and the Hydrogen Seal Oil unit is at 731' elevation, but incorrect because these actions are not required unless the fire header pressure cannot be maintained above 125 psig and given fire header pressure is at 130 psig.

Distractor 3 is plausible because LOA-FP-101 directs removing the Main Generator from service to throttle service water cooled components to maintain fire header pressure, but incorrect because these actions are not required unless fire header pressure cannot be maintained above 125 psig and the given fire header pressure is 130 psig.

Reference: LOA-FP-101 UNIT 1 FIRE PROTECTION SYSTEM ABNORMAL, Rev 47

Reference provided during examination: None

Cognitive level: High
10CFR Part 55: 41.5 / 41.10

Level (RO/SRO): 3.3
Tier: 1 Group: 1

PRA: No
K/A: 600000 AK3.02 - Knowledge of the reasons for the following responses or actions as they apply to (APE 24) PLANT FIRE ON SITE: Steps called out in the site fire protection plant, fire protection system manual, and fire zone manual

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Safety Function: 8
SRO Justification: N/A

Question Source: New
Question History:

Comments:
Objective: 125.00.15

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

20

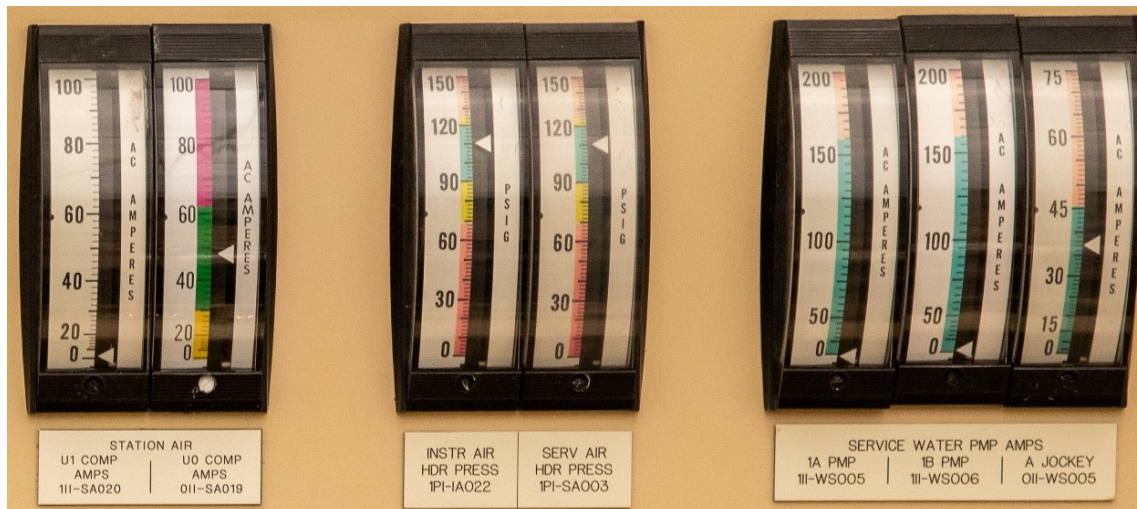
ID: 2543505

Points: 1.00

Unit 1 is at 100% RTP during summer conditions.

- Emergency Load Conservation measures are in place
- Grid voltage is projected to drop to 348.5 kV due to a trip of another power plant
- LOA-GRID-001, Low Grid Voltage, Tables 1 and 2 are met and Minimum Allowable Switchyard voltage is 350.7 kV
- 3 Circulating Water pumps are operating
- 3 Heater Drain Pumps are operating
- C VP chiller is secured

Using the following indications and Table 3 from LOA-GRID-001, determine the Minimum Allowable Switchyard voltage.



Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Unit 1 Table 3		
Equipment	Number Running assumed in ETAP Model	Required Configuration to allow Decrease in Minimum Allowable Switchyard Voltage
Circ Water Pumps	3 (1A/1B/1C)	0.4 kv for each CW pump less than 3 that are OFF.
Station Air Compressors	2 (U0 and U1)	0.4 kv for each U1 powered SAC less than 2 that is OFF.
Service Wtr Pumps	2 (1A/1B)	1.0 kv for each U1 powered WS pump less than 2 that is OFF. • One (1) U1 powered WS pump running a 1.0 kv deduction is allowed. • Zero (0) U1 powered WS pump running a 2.0 kv deduction is allowed.
Htr Drain Pumps	3	1.4 kv for each HD pump less than 3 that are OFF.
C VP Chiller	1	0.4 kv when 1C VP chiller is OFF.

- A. 347.5 kV
- B. 347.9 kV
- C. 348.3 kV
- D. 348.9 kV

Answer: B

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "347.9 kV" is the correct answer.

Per LOA-GRID-001

Reductions in allowable switchyard voltage based on equipment configuration = 0.4 (U1 SA compressor) + 2.0 (1A and 1B WS pumps) + 0.4 (C VP chiller) = 2.8 kV

Minimum allowable switchyard voltage = 350.7 kV – 2.8 kV = 347.9 kV

Distractor 1 is plausible because operation with 2 CW pumps is a common configuration, but incorrect because during summer operation there are normally 3 Circ water pumps in operation.

Distractor 2 is plausible because lake temperatures are normally elevated in the summer and operation of the C VP chiller in addition to the A or B chiller is an allowed configuration, but incorrect because C VP is not normally operated in the summer.

Distractor 3 is plausible because an operating service water jockey pump could be mistaken for an operating service water pump, but incorrect because the table only credits the main service water pumps in evaluation of minimum switchyard voltage.

Reference: LOA-GRID-001, Rev 17

Reference provided during examination: Embedded photo of instrument gauges and LOA-GRID-001 Table 3

Cognitive level: High

10CFR Part 55: 41.10

Importance Level (RO/SRO): 3.9 / 3.8

Tier: 1 Group: 1

PRA: Yes

K/A: 700000 G2.1.19 - (APE 25) GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES, CONDUCT OF OPERATIONS Ability to use available indications to evaluate system or component status

Safety Function: 6 - Electrical

SRO Justification: N/A

Question Source: New

Question History: N/A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Comments:

Objective #: 003.00.15

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

21

ID: 2543685

Points: 1.00

Unit 1 is at 5% RTP with the Mechanical Vacuum Pump in operation.

- The steam jet air ejectors are placed in operation
- A and B Off-gas (OG) Post Treatment Radiation High High alarms
- C and D Main Steam Line (MSL) Rad Monitor High alarms
- C and D Main Steam Line (MSL) Rad Monitor High High are NOT in alarm
- Control Switch for 1N62-F057, OG Vent Stack System Isolation valve, is in the OPEN position
- Control switches for 1N62-F300A and B, Main Condenser OG outlet valves, are in the AUTO position

What is the required action and why?

- A. Close 1N62-F057, immediately, because OG post treat radiation levels are above the isolation required value and 1N62-F057 will not close automatically.
- B. Close 1N62-F300A and B in 15 minutes, because OG post treat radiation levels are above the isolation required setpoint and 1N62-F300A and B will not close automatically.
- C. Shutdown the Mechanical Vacuum Pump in 15 minutes, because MSL radiation level are above the isolation required value and the MVP will not shutdown automatically.
- D. Scram and close the Main Steam Isolation Valves (MSIVs) immediately, because MSL radiation levels are above the isolation required value and the MSIVs will not close automatically.

Answer: A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Close 1N62-F057, immediately, because OG post treat radiation levels are above the isolation required value and 1N62-F057 will not close automatically." is the correct answer.

The Off Gas Pretreatment Monitoring system provides alarms when the OG System influent radiation level approaches the ODCM (refer to REC 12.2.2). Per LOR-1N62-P600-B207, Off Gas System Post Treatment Radiation High, a Hi-Hi-Hi rad condition will result in isolation of 1N62-F057 Off Gas Vent Stack System isolation valve. With the 1N62-F057 control switch in OPEN the isolation signal is bypassed requiring the operator to place the control switch in AUTO or CLOSE to isolate off-gas post treatment.

LOR-1N62-P600-B207
Revision 10
November 25, 2022

Description: Off Gas Post Treatment Radiation High

Computer Print: R1277 OG Post-Trtmt Rad Hi
R1278 OG Post-Trtmt Rad Hi-Hi-Hi
R1279 OG Post-Trtmt Rad Hi-Hi

Setpoint: Multiple

Sensor No: Multiple

Alarm No: 1PR27A/B/C

Drawing: 1E-1-4218AG

Activating Device: 1D18A-Z3B
1D18-R601

OFF GAS
POST-TRTMT
RAD HI

A. AUTOMATIC ACTIONS

NOTE

Automatic actions occur if both channels are in the trip condition.

1. Hi - (If Charcoal Adsorber Train Mode Switch is in AUTO position ONLY).
 - a. CLOSES 1N62-F043, Off Gas Charcoal Adsorber Bypass Valve.
 - b. OPENS 1N62-F042, Off Gas Charcoal Adsorber Inlet Valve.
2. Hi-Hi - NO AUTOMATIC ACTIONS. Alarm comes off of recorder 1D18-R601 at panel 1H13-P600. This is intended to be a warning prior to receiving OG Isolation/Hi-Hi-Hi alarm.
3. Hi-Hi-Hi - ISOLATES Off Gas System by Closure of 1N62-F057, Off Gas System to Stack Discharge Valve, 1N62-F085A/B, Holdup Volume Drain Valve and 1N62-F030A/B, Cooler Condenser Drain Valve.

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

12.2.2 Radioactive Gaseous Effluent Monitoring Instrumentation

REC 12.2.2 The Radioactive Gaseous Effluent Instrumentation channels in Table R12.2.2-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of REC 12.4.1 are not exceeded.

CY-LA-170-30

Revision 1

August 202

Part I, Radiological Effluent Control

Table R12.2.2-1 (page 1 of 2)
Radioactive Gaseous Effluent Monitoring Instrumentation

INSTRUMENT ^(a)	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER INSTRUMENT	CONDITION REFERENCED FROM REQUIRED ACTION A.2 AND B.1	SURVEILLANCE REQUIREMENTS
1. Main Condenser Offgas Treatment System Effluent Monitoring System				
a. Noble Gas Activity Monitor – Providing Alarm and Automatic Termination of Release (Post-Treat)	(b)	2	C, if only one required channel inoperable D, if both required channels inoperable	RSR 12.2.2.1 RSR 12.2.2.2 RSR 12.2.2.5 RSR 12.2.2.7 ^(e)

2.5.4 Condenser Air Ejector Monitors

Monitors 1(2)D18-N002/N012 (pre-treatment) and 1(2)D18-N903A/B (post-treatment) continuously monitor gross gamma activity downstream of the steam jet air ejector and prior to release to the main stack.

On a "high-high-high" alarm, monitor 1(2)D18-N903A/B automatically initiates closure of valve 1(2)N62-F057 terminating the associated offgas release.

11.5.2.1.2 Off-Gas Posttreatment Radiation Monitor

This monitor subsystem measures the radioactivity in the off-gas vent pipe after the normal off-gas has been processed through the off-gas treatment (RECHAR) system and before it discharges into the elevated release duct. This complies with General Design Criteria 13, 60, 63, and 64.

Distractor 1 is plausible because the Main Condenser OG outlet valves are closed when MSL radiation levels are high enough, but incorrect because these valves will isolate automatically when OG flow is stopped.

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Distractor 2 is plausible because the Mechanical vacuum pump must be secured if the MSL radiation levels are too high to minimize the radioactive release, but incorrect because this is required when the MSL Hi-Hi alarm is received.

Distractor 3 is plausible because this is the required action if release rates are high enough, but incorrect because offsite release rates are determined from WRGM indications not MSL radiation levels.

Reference: LOR-1N62-P600-B207 Off Gas Post Treatment Radiation High, rev 10; LaSalle ODCM, CY-LA-170-301 rev 11; UFSAR 11.5.2; LOR-1H13-P601-E402 rev 3; LOR-1H13-P601-E403 rev 8; 1E-1-4229AQ rev D;
Reference provided during examination: None

Cognitive level: High
10CFR Part 55:41.5 / 41.10

Importance Level (RO/SRO): 4.0
Tier: 1 Group: 2

PRA: No
K/A: 295017 AK3.01 - Knowledge of the reasons for the following responses or actions as they apply to (APE 17) ABNORMAL OFFSITE RELEASE RATE:
System isolations

Safety Function: 9 – Rad Release
SRO Justification: N/A

Question Source: New
Question History: N/A

Comments:

Objective #052.00.04

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

22

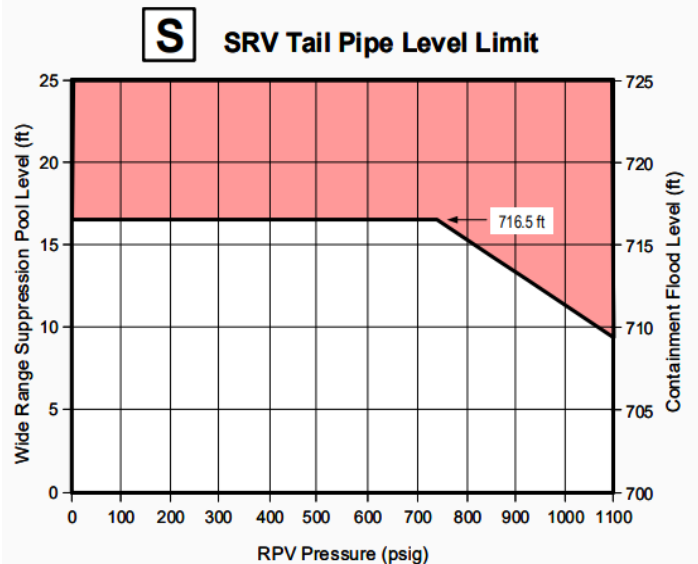
ID: 2543698

Points: 1.00

Unit 2 was at 100% RTP when a LOCA occurred.

- RPV pressure is 300 psig and lowering 1 psig/min
- RPV level is -12 inches and rising 3 in/min
- Suppression pool water level is 716 ft rising 0.05 ft/min
- RCIC is injecting at 600 gpm
- RCIC suction cannot be aligned to the suppression pool
- HPCS is injecting at 6000 gpm

What pumps will be kept injecting, if any?



- A. HPCS Only
- B. RCIC Only
- C. HPCS and RCIC
- D. Neither HPCS nor RCIC

Answer: A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

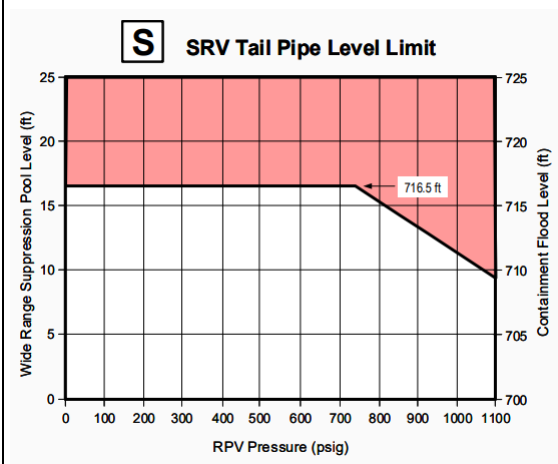
Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "HPCS only" is the correct answer.

Per LGA-003 SP/L-5, if suppression pool level and RPV pressure cannot be held below SRV tailpipe level limit, Fig. S, then stop injection into the RPV from outside the primary containment not needed for core cooling or to shutdown the reactor. With suppression pool water level continuing to rise at a rate consistent with the current RCIC injection rate, securing injection from RCIC will halt the rising trend in suppression pool water level. HPCS will be used to inject since it is taking suction from inside primary containment



Distractor 1 is plausible because RCIC is supplying makeup water to the RPV, but incorrect because injection sources external to primary containment need to be secured due to approaching the USAFE side of the SRV tailpipe limit.

Distractor 2 is plausible because HPCS is injecting from the suppression pool, but incorrect because RCIC is injecting from an external water source.

Distractor 3 is plausible because RCIC normal suction is from the CST and is required to be secured and HPCS has piping connecting the CST, but incorrect because the HPCS piping to the CST has been abandoned and HPCS suction is from the suppression pool.

Reference: LOP-RH-16 rev 27, LGA-003 rev 19, LGA figure Y, NC, and S
Reference provided during examination: LGA figure S "SRV tail pipe level limit"

Cognitive level: High
10CFR Part 55: 41.8 / 41.10

Importance Level (RO/SRO): 3.3
Tier: 1 Group: 2

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

PRA: yes

K/A: 295029 EK2.03 - Knowledge of the relationship between the (EPE 6) HIGH SUPPRESSION POOL WATER LEVEL and the following systems or components: HPCS

Safety Function: 4

SRO Justification: N/A

Question Source: New

Question History: N/A

Comments:

Objective #: 503.00.08

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

23

ID: 2543708

Points: 1.00

Unit 1 was at 100% RTP when a seismic event resulting in a LOCA occurs.

- Reactor Scram occurred with all rods in
- Bus 141Y tripped on overcurrent
- RPV pressure is 800 psig and lowering 1 psig/min
- RPV level dropped to -70 inches and is rising at 1 inch/min
- RB RCIC/LPCS room radiation level is 2000 mRem/hr and rising 100 mRem/hr per minute
- RB RCIC room ARM is in alarm
- RB RCIC/LPCS room temperature is 140°F and rising 10°F/min
- RB North HCU radiation level is 580 mRem/hr and rising 15 mRem/hr per minute
- RB South HCU radiation level is 500 mRem/hr and rising 10 mRem/hr per minute

Per LGA-002, Secondary Containment Control (1-3), what is the next operator action?

- A. Secure RCIC and isolate RCIC pump
- B. Secure LPCS and isolate LPCS pump
- C. Secure RCIC and isolate RCIC turbine
- D. Remove fuses for SDV vents and drains

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Secure RCIC and isolate turbine" is the correct answer.

RCIC turbine has a steam leak as indicated by elevated rad levels and room temperature.

Per LGA-002,

If any area temperature or radiation is above max normal (Details R, T), then

Isolate all discharges into affected areas except systems needed for:

- o Damage control
- o LGA actions (venting, reactor shutdown, etc.)

Distractor 1 is plausible because the RCIC pump is located in the room, but incorrect because a breach of RCIC water piping would result in room flooding and there would not be high temperature or radiation due to supply from the CST.

Distractor 2 is plausible because the LPCS pump is located in the room, but incorrect because a breach of LPCS water piping would result in room flooding and the LPCS pump is not running due to loss of bus 141Y.

Distractor 3 is plausible because failure of SDV vents and drains to close would result in reactor water input into the RBEDT, but incorrect because there is no flooding and RPS buses successfully completed the scram which depressurized the scram air header including the actuators for the SDV vents and drains which close on loss of air pressure.

Reference: LGA-002, Rev 11; LOR-1H13-P601-C408, Rev 4; LOR-1PM01J-A214, Rev 1

Reference provided during examination: None

Cognitive level: High

10CFR Part 55: 41.10

Importance Level (RO/SRO): 4.0 / 3.8

Tier: 1 Group: 2

PRA: No

K/A: 295033 EA2.03 - Ability to determine or interpret the following as they apply to (EPE 10) HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Source of high area radiation

Safety Function: 9 – Rad Release

SRO Justification: N/A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Question Source: New

Question History: N/A

Comments:

Objective #: 052.00.11

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

24

ID: 2505194

Points: 1.00

Units 1 and 2 are at 100% RTP.

On Unit 2:

- Annunciator 2H13-P601-F205, Div 1 Fuel Pool Rad Hi-Hi, alarms
- Fuel Pool Radiation Monitors 2A and 2B are tripped

Two minutes later:

On Unit 1:

- Annunciator 1H13-P601-E205, Div 2 Fuel Pool Rad Hi-Hi, alarms
- Fuel Pool Radiation Monitors 1C and 1D are tripped

The crew will be directed to enter (1)_____ to verify isolations are complete to prevent (2)_____.

- A. (1) ONLY the Unit 2 LGA;
(2) off-site doses from exceeding 10 CFR limits
- B. (1) ONLY the Unit 2 LGA;
(2) over exposure of personnel on the refuel floor
- C. (1) BOTH Unit 1 and Unit 2 LGAs;
(2) off-site doses from exceeding 10 CFR limits
- D. (1) BOTH Unit 1 and Unit 2 LGAs;
(2) over exposure of personnel on the refuel floor

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "(1) both Unit 1 and Unit 2 LGAs; (2) off-site doses from exceeding 10 CFR limits" is correct.

Per LOR-2H13-P601-F205 and LOR-1H13-P601-E205, the operators are directed to enter LGAs for both units because the high radiation trip set point has been exceeded for both units even if all system isolations and actuations for both units occur with the first actuation. The crews will verify isolations are complete because the TS Bases states that the VR and Fuel Pool Radiation Monitor trip setpoints are based on NOT exceeding 10CFR20 and 10CFR100 limits.

Per LOR-2H13-P601-F205, LOR-1H13-P601-E205

LOR-2H13-P601-F205
Revision 6
July 26, 2022

Description: Div I Fuel Pool Radiation High High

Computer Print: R0484 Div I Fuel Pool Rad Hi-Hi

Setpoint: Actual: 10 mR/hr

Tech Spec: 42 mR/hr

Sensor No: 2D18-N015A/B

Alarm No: 2PR36A

Drawing: 1E-2-4218AC

Activating Device: 2D18-K615A/B

DIV 1
FUEL POOL
RAD HI-HI

A. AUTOMATIC ACTIONS

NOTE

Actions occur if BOTH Channels "A" and "B" have tripped.

1. CLOSES Outboard Isol. Dampers and TRIPS U1 and U2 Reactor Building Ventilation System.
2. STARTS both U1 and U2 Standby Gas Treatment System Trains.
3. CLOSES Primary Containment Vent and Purge Outboard Isolation Valves.

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

LOR-1H13-P601-E205
Revision 5
June 26, 2022

Description: Div II Fuel Pool Radiation High High
Computer Print: R1291 Div II FUEL POOL RAD HI-HI
Setpoint: Actual: 8 mR/hr
Tech Spec: 42 mR/hr
Sensor No: 1D18-N015C/D
Alarm No: 1PR37A
Drawing: 1E-1-4218AC
Activating Device: 1D18-K615CX/DX

DIV 2
FUEL POOL
RAD HI-HI

A. AUTOMATIC ACTIONS

NOTE

Actions occur if both Channels C and D have tripped.

1. CLOSES Inboard Isol Dampers and TRIPS U1 and U2 Reactor Building Ventilation System.
2. STARTS Both U1 and U2 Standby Gas Treatment System Trains.
3. CLOSES Primary Containment Vent and Purge Inboard Isolation Valves.

B. OPERATOR ACTIONS

1. CHECK Channel C and D monitors at backpanel 1H13-P636 to see if both or only one channel has tripped. A single channel trip may be indicative of a failed channel and no radiation problems exists.
2. VERIFY automatic actions occur if appropriate. If SGBT Trains have auto-initiated, REFER to LOP-VG-03.
3. ENTER LGA-002 secondary containment control, as appropriate, if actuation was result of a valid radiological condition.

Per TS Bases 3.3.6.2

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

The secondary containment isolation instrumentation automatically initiates closure of appropriate secondary containment isolation valves (SCIVs) and starts the Standby Gas Treatment (SGT) System. The function of these systems, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Refs. 1 and 2), such that offsite radiation exposures are maintained within the requirements of 10 CFR 100 that are part of the NRC staff approved licensing basis. Secondary containment

Distractor 1 is plausible because Unit 2 trips will complete all actuations and isolations for both units, but incorrect because Unit 1 trips meet the entry conditions for Unit 1 such that LGA entry for both units is required.

Distractor 2 is plausible because Unit 2 trips will complete all actuations and isolations for both units and high radiation on the refuel floor will affect refueling floor personnel, but incorrect because Unit 1 trips meet the entry conditions for Unit 1 such that LGA entry for both units is required and FC/VR ventilation isolation will not reduce refuel floor dose rates.

Distractor 3 is plausible because high radiation on the refuel floor will affect refueling floor personnel, but incorrect because FC/VR ventilation isolation will not reduce refuel floor dose rates.

Reference: LOR-2H13-P601-F205, Rev 6; LOR-1H13-P601-E205, Rev 5; TS bases 3.3.6.2

Reference provided during examination: None

Cognitive level: High

10CFR Part 55: 41.5 / 41.7

Importance Level (RO/SRO): 3.9

Tier: 1 Group: 2

PRA: No

K/A: 295034 EK1.02 - Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the (EPE 11) SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: Radiation releases

Safety Function: 9 - Rad Release

SRO Justification: N/A

Question Source: Modified

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Question History: N/A

Comments: from question 1253772

Objective #: 118.00.12

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

25

ID: 2544000

Points: 1.00

Unit 1 is at 100% RTP with Unit 2 Reactor Building ventilation (VR) OOS for maintenance.

- Power to 1VR04YA and 1VR05YB Secondary Containment Isolation valve solenoids fails
- Secondary containment pressure is 0.0 Inches of H₂O and stable

What action will restore Secondary Containment dP to an acceptable band?

- A. Manually start Unit 1 SBGT
- B. Start standby VR exhaust fan ONLY
- C. Secure unnecessary VR supply fans ONLY
- D. Balance the number of running VR exhaust and supply fans

Answer: A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

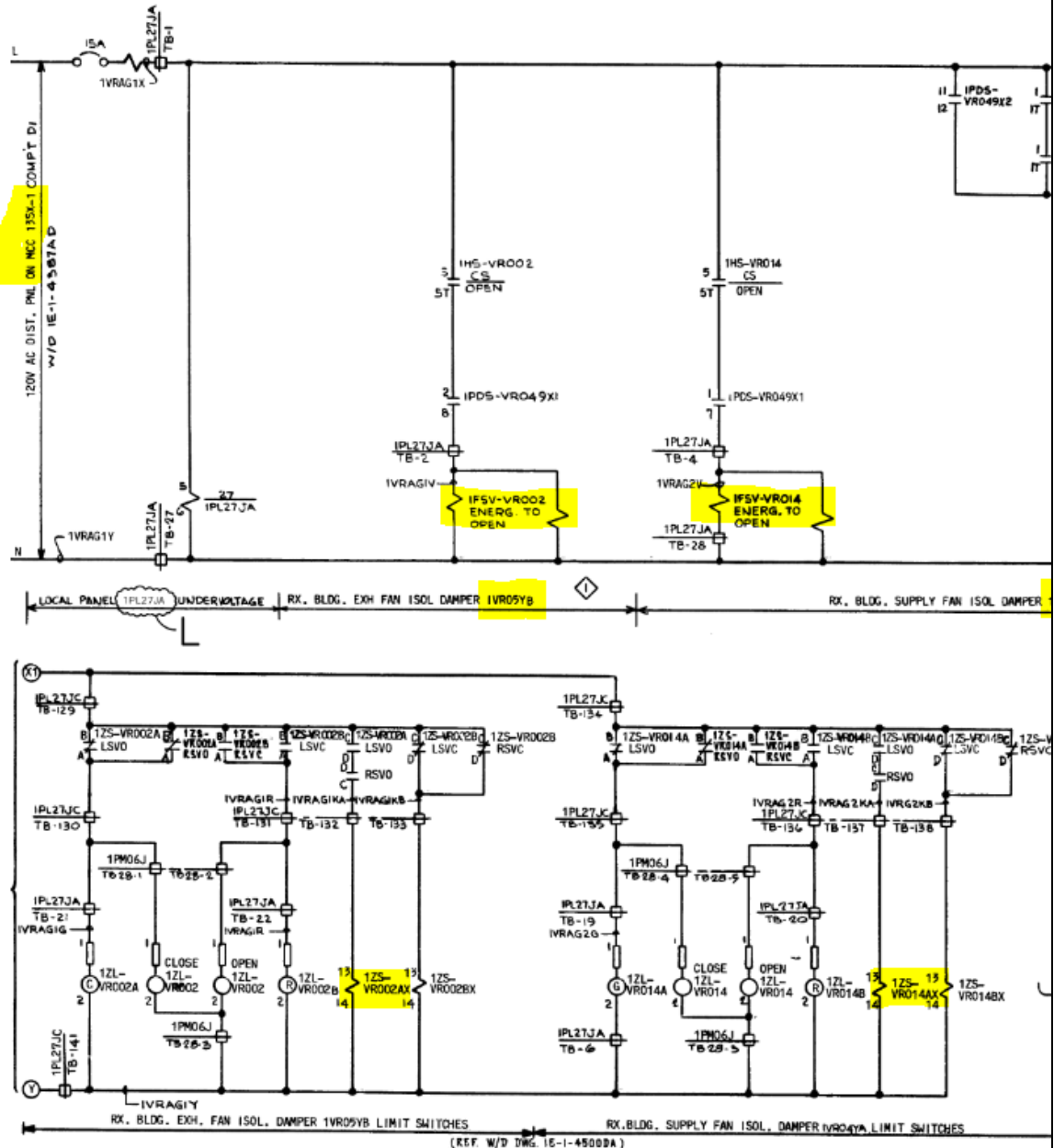
Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: The correct answer is "Manually start Unit 1 SBT" Per Dwg 1E-1-4083AA, AB, AC and AG the loss of power to the solenoids will cause the valves to fail closed. When the isolation valves are closed the VR supply and exhaust fans will trip and RB dP will rise to 0 Inches H₂O. Per LOA-VR-101 the next action is to manually start the Unit 1 SBT train.

Per 1E-1-4083AG



Test Answer Key

LaSalle ILT 23-1 NRC RO

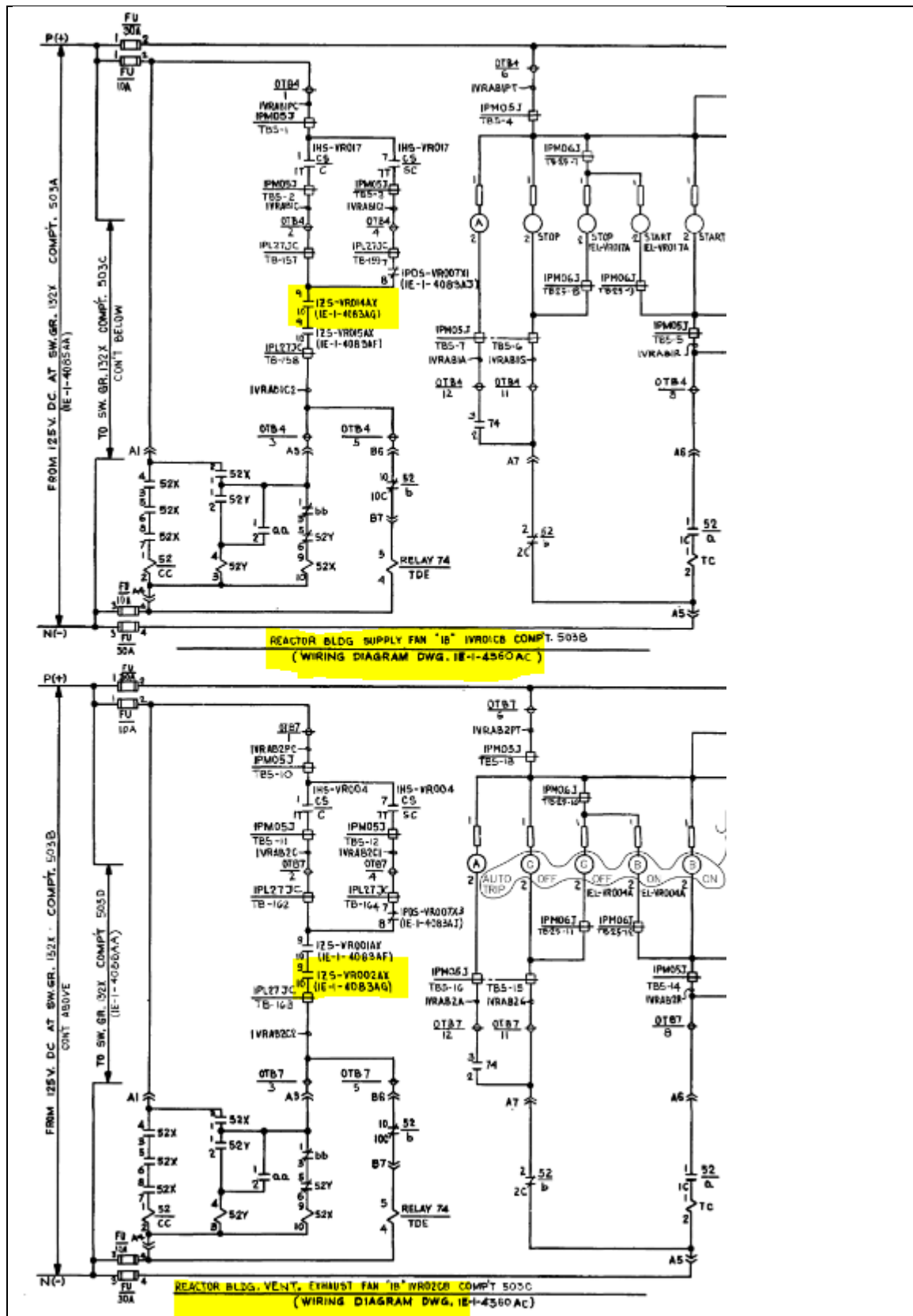
Test ID: 375864

Per 1E-1-4083AB (1B Fan Logic typical of 1A and 1C VR fans)

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864



Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Distractor 1 is plausible because starting a VR exhaust fan would make the Reactor Building dP more negative, but incorrect as none of the Exhaust fans can be started with the Div 1 secondary containment isolation valves closed.

Distractor 2 is plausible because securing VR supply fans will drive Reactor building DP more negative, but incorrect because supply fan operation will not impact reactor building DP with the Division 1 secondary containment isolation valves closed.

Distractor 3 is plausible because matching the number of supply and exhaust fans is a normal LOR action if there is less than 2 exhaust or supply fans running, but incorrect because none of the fans will run with the Div 1 secondary containment isolation valves closed.

Reference: LOA-VR-101, UNIT 1 RECOVERY FROM A GROUP 4 ISOLATION OR SPURIOUS TRIP OF REACTOR BUILDING VENTILATION, rev 16; Dwg 1E-1-4083AA rev M, 1E-1-4083AB rev J, 1E-1-4083AC rev H and 1E-1-4083AG rev L

Reference provided during examination: None

Cognitive level: High
10CFR Part 55: 41.5 / 41.7

Importance Level (RO/SRO): 3.8
Tier: 1 Group: 2

PRA: No
K/A: 295035 EA1.02 - Ability to operate or monitor the following as they apply to (EPE 12) SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: SGTs/FRVS

Safety Function: 9 – Rad Release

SRO Justification: N/A

Question Source: New

Question History: N/A

Comments:

Objective: 118.00.16 DERIVE VR response to component failures.

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

26

ID: 2544028

Points: 1.00

Unit 1 was at 100% RTP when a LOCA occurred.

- RPV Pressure is 45 psig and steady
- Drywell Pressure is 18 psig and steady
- Suppression Chamber pressure is 13 psig and steady
- Drywell hydrogen concentration is 3% and rising
- Drywell oxygen concentration is 2.5% and rising

LGA-HG-101, Operation of the Hydrogen Recombiner as a Mixing System, is entered.

The Unit 1 Hydrogen Recombiner ...

- A. must be started to ensure that hydrogen does NOT accumulate in the Drywell Head or Reactor Pedestal Areas.
- B. must be started as a mixing system taking a suction from the Drywell and discharging to the Suppression Chamber.
- C. can NOT be started because the combination of hydrogen and oxygen in the Drywell is greater than the explosive limit.
- D. can NOT be started because running the recombiner would create a direct pathway between the Drywell and Reactor Building.

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "can NOT be started because running the recombiner would create a direct pathway between the Drywell and Reactor Building." is the correct answer.

The hydrogen recombiner is interlocked with high Drywell Pressure of 15.3 psig which is above the LGA-003 entry condition of 1.97 psig to prevent exceeding design pressure of the hydrogen recombiner system.

The hydrogen recombiner is operated when Drywell Hydrogen is above 3% prior to venting the drywell to prevent hydrogen accumulation in upper drywell areas.

Per LGA-HG-101 Section D "Limitations": "The discharge line loop seal will blow out at Drywell pressures exceeding 15.3 psig, creating a pathway between the Drywell and Reactor Building."

D. LIMITATIONS

- _____ 1. High inlet pressure greater than 30 psia (15.3 psig) will trip the Hydrogen Recombiner blower and close the inlet valve (FV-1). It may be necessary to use containment systems to reduce the pressure before the blower can be restarted again.
- _____ 2. The discharge line loop seal will blow out at Drywell pressures exceeding 15.3 psig, creating a pathway between the Drywell and the Reactor Building.
(NTS 373-100-93-01627)

Distractor 1 is plausible because the H₂ recombiner is used for mixing the DW atmosphere, but incorrect because drywell pressure is in excess of the capability of the loop seal (15.3 psig) and the recombiner will trip and isolate at the given DW pressure.

Distractor 2 is plausible because the H₂ will concentrate in those areas due to low natural convection flow, but incorrect because drywell pressure is in excess of the capability of the loop seal (15.3 psig) and the recombiner will trip and isolate at the given DW pressure.

Distractor 3 is plausible because high H₂ And O₂ levels are an ignition source that preclude recombiner operation per LGA-HG-101, but incorrect because H₂ level is NOT above the explosive limit (4%)

Reference: LGA-HG-101, Rev 3

Reference provided during examination: None

Cognitive level: High
10CFR Part 55: 41.7

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Importance Level (RO/SRO): 4.5

Tier: 1 Group: 2

PRA: No

K/A: 500000 G2.4.2 - HIGH CONTAINMENT HYDROGEN CONCENTRATION (EPE 16) - EMERGENCY PROCEDURES / PLAN Knowledge of system setpoints, interlocks and automatic actions associated with emergency and abnormal operating procedure entry conditions

Safety Function: 5 - Containment integrity

SRO Justification: N/A

Question Source: Modified (copied from 1260649)

Question History: 2008 NRC exam

Comments: Modified to make appropriate for the RO license level from SRO level question

Objective #: 094.00.10

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

27

ID: 2544050

Points: 1.00

Unit 2 is at 100% RTP.

- Surveillance LOS-RH-Q1 Attachment 2A, RHR System Operability and Inservice Test, completed for the 2A RHR pump
- Pump discharge pressure is 135 psig (TS acceptance criteria > 130 psig)
- Maximum Pump flow rate is 7000 gpm (TS acceptance criteria > 7200 gpm)

What is the status of the 2A RHR pump?

What action is required?

- A. 2A RHR is INOPERABLE; place protected pathways around redundant equipment
- B. 2A RHR is DEGRADED; operability will be determined by engineering analysis of test results
- C. 2A RHR is in the ALERT range; place protected pathways around redundant equipment
- D. 2A RHR is in the ALERT range; operability will be determined by engineering analysis of test results

Answer: A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "2A RHR is INOPERABLE; place protected pathways around redundant equipment" is the correct answer.

Since the pump flow rate is less than the TS acceptance criteria, which is the required action range, the pump must be declared inoperable and protected path barriers established to prevent damage to redundant equipment.

Per LOS-RH-Q1: If pump test data falls in the REQUIRED ACTION range, ER-AA-321 is used to determine applicable actions.

D.1 If pump or valve data fall into the ALERT or REQUIRED ACTION range, ER-AA-321, Administrative Requirements for Inservice Testing, shall be referred to for applicable actions.

D.3 The required Technical Specification RHR flowrate for LPCI injection and for containment cooling is equal to or greater than 7200 gpm when tested pursuant to LSCS In-Service Testing Program for Pumps and Valves. UFSAR Table 6.2-2 requires a flow equal to or greater than 7450 gpm for containment cooling. The RHR pumps are tested to the higher flow for containment cooling.

Per ER-AA-321: A pump not being tested as part of a post maintenance testing that falls in the required action range requires the pump to be declared INOPERABLE

Per OP-AA-108-117: "When SSCs are planned to or become unavailable, then PROTECT redundant equipment if plant configuration is such that redundant equipment unavailability or manipulation would cause:...

An entry into Tech Spec 3.0.3 (3.0.1 for TMI) or a shutdown Tech Spec LCO of 12 hrs or less (i.e. be in hot shutdown in 12 hrs or less)."

Distractor 1 is plausible because 2A RHR pump characteristics are DEGRADED and normally degraded pumps will require an evaluation to determine if the pump will still meet its safety function if the testing values have deviated from the pump characteristics curve, but incorrect because RHR flow requirements are based on UFSAR accident analysis and not derived from the RHR pump characteristics curve.

Distractor 2 is plausible because protected pathways are required to be placed around redundant safety equipment, but incorrect because 2A RHR pump surveillance data is below the TS required value, the REQUIRED ACTION range, and the RHR pump is INOPERABLE.

Distractor 3 is plausible because a component in the ALERT range requires an operability determination based on test results, but incorrect because the 2A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

RHR pump surveillance data is below the TS required value, the REQUIRED ACTION range, and the RHR pump is INOPERABLE.

Reference: LOS-RH-Q1, Rev 101; OP-AA-108-117, Rev 7; ER-AA-321, Rev 14, ITS 3.5.1

Reference provided during examination: None

Cognitive level: High

10CFR Part 55: 41.5

Importance Level (RO/SRO): 3.0 / 3.3

Tier: 2 Group: 1

PRA: No

K/A: 203000 A2.20 - Ability to (a) predict the impacts of the following on the (SF2, SF4 RHR/LPCI) RHR/LPCI: INJECTION MODE and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Surveillance acceptance criteria not being met

Safety Function: 2 – Inventory control

SRO Justification: N/A

Question Source: New

Question History: N/A

Comments:

Objective #: 64.00.16

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

28

ID: 2504707

Points: 1.00

Unit 1 was at 100% RTP when a loss of offsite power occurred.

Drywell pressure is 2.1 psig and rising.

What is the Unit 1 Residual Heat Removal (RHR) pump starting sequence when their associated buses are re-energized by their respective Emergency Diesel Generators?

The ____ (1) ____ will start immediately and then the ____ (2) ____ will start after a 5 sec time delay.

- A. (1) 1A and 1C pump;
(2) 1B pump
- B. (1) 1C pump;
(2) 1A and 1B pumps
- C. (1) 1B pump;
(2) 1A and 1C pumps
- D. (2) 1A pump;
(2) 1B and 1C pumps

Answer: B

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: “(1) 1C pump; (2) 1A and 1B pumps“ is the correct answer.

Per Tech Spec 3.3.5.1 Table 3.3.5.1-1

The 1(2)C RHR pumps and the LPCS pumps start immediately after the vital bus is reenergized by the Diesel Generator. To prevent overloading the Diesels with excessive starting currents, the 1(2)A and the 1(2)B RHR pumps will start after a 5-second time delay. By this time the starting current from the 1(2)C RHR pump and the LPCS pump should have died-off thereby preventing the diesels from becoming overloaded.

Table 3.3.5.1-1 (page 1 of 4)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWA VALU
c. LPCI Pump A Start-Time Delay Relay	1,2,3, 4 ^(a) ,5 ^(a)	1	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 5.5 s
c. LPCI Pump B Start-Time Delay Relay	1,2,3, 4 ^(a) ,5 ^(a)	1	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 5.5 s

Distractor 1 is plausible if the 1A and 1C RHR Pumps start first, but incorrect as only the 1C RHR pump starts first.

Distractor 2 is plausible if the 1B RHR Pump starts first, but incorrect as only the 1C RHR pump starts first.

Distractor 3 is plausible if the 1A RHR Pump starts first, but incorrect as only the 1C RHR pump starts first.

Reference: Tech Spec 3.3.5.1 Table 3.3.5.1-1

Reference provided during examination: none

Cognitive level: Memory
10CFR Part 55: CFR: 41.7

Level (RO/SRO): 3.8
Tier: 2 Group: 1

PRA: No

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

K/A: 203000: K4.07 RHR/LPCI: Injection Mode: Knowledge of RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) design feature(s) and/or interlocks which provide for the following: Emergency generator load sequencing

Safety Function: 2

SRO Justification: N/A

Question Source: Bank

Question History:

Comments:

Objective # 064.00.07

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

29

ID: 2544229

Points: 1.00

Unit 2 is in Mode 4.

- RPV temperature is 195°F and steady
- 2A RHR pump is in shutdown cooling
- 2A and 2B RHR Service Water (RHR WS) pumps are in operation

Subsequently:

- Annunciator 2H13-P601-C202, 2A RHR SERV WTR STRAINER DP HI, alarms
- All strainer backwashing CANNOT restore the RHR WS strainer dP to the normal band
- 2A RHR WS strainer dP is 16 psid rising at 1 psid/min

Which of the following indications will be observed FIRST?

- A. 2A RHR WS pump cavitation
- B. 2B RHR WS Pump motor amps Hi
- C. 2B RHR pump seal temperature Hi
- D. 2A RHR HX outlet temperature rising

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "2A RHR HX inlet temperature rising" is the correct answer.

The fouled strainer which is located downstream of the 2A and 2B RHR WS pumps will result in reduced service water flow to the RHR heat exchanger and a temperature increase during a high heat load. As RPV temperature was steady with current plant conditions prior to the service water strainer DP rising, the reduction in cooling water flow will result in a rising temperature in the RPV which will result in a rising temperature on the inlet side of the RHR heat exchanger.

Distractor 1 is plausible because a loss of suction to a pump will cause cavitation such as with the non-safety service water pumps, but incorrect because the strainer is downstream of the RHR SW pumps which will not result in any reduction in NPSH.

Distractor 2 is plausible because high motor amps is an indication of a pump flow concern, but incorrect because high amps is indicative a pump in runout with excessive flow not reduced flow which will be indicated by lower than normal amps.

Distractor 3 is plausible because the RHR pump coolers are supplied by RHR SW, but incorrect because the 2B RHR pump is supplied by Div 2 'C' and 'D' RHR Service water pumps.

Reference: Dwg M-134, Sheet 2; GEFES BWR / COMPONENTS / CHAPTER 2 / PUMPS, Rev 4c

Reference provided during examination: None

Cognitive level: High
10CFR Part 55: 41.7

Importance Level (RO/SRO): 3.3
Tier: 2 Group: 1

PRA: No
K/A: 205000 K6.08 - Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF4 SCS) SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE):
Service water

Safety Function: 4
SRO Justification: N/A

Question Source: New

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Question History: N/A

Comments:

Objective #: 064.00.13a

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

30

ID: 2504713

Points: 1.00

Unit 1 was at 100% RTP when a spurious Division 1 LPCS Initiation signal occurred and cleared.

The Unit 1 Division 1 ECCS Initiation Logic Reset pushbutton is located at (1)_____, and depressing it will (2)_____.

- A. (1) 1H13-P603
(2) reset the RCIC system initiation logic
- B. (1) 1H13-P601
(2) reset the RHR and LPCS system initiation logic
- C. (1) 1H13-P603
(2) cause the RHR and LPCS pumps to automatically stop
- D. (1) 1H13-P601
(2) automatically close the RHR and LPCS injection valves and open the respective minimum flow valve

Answer: B

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

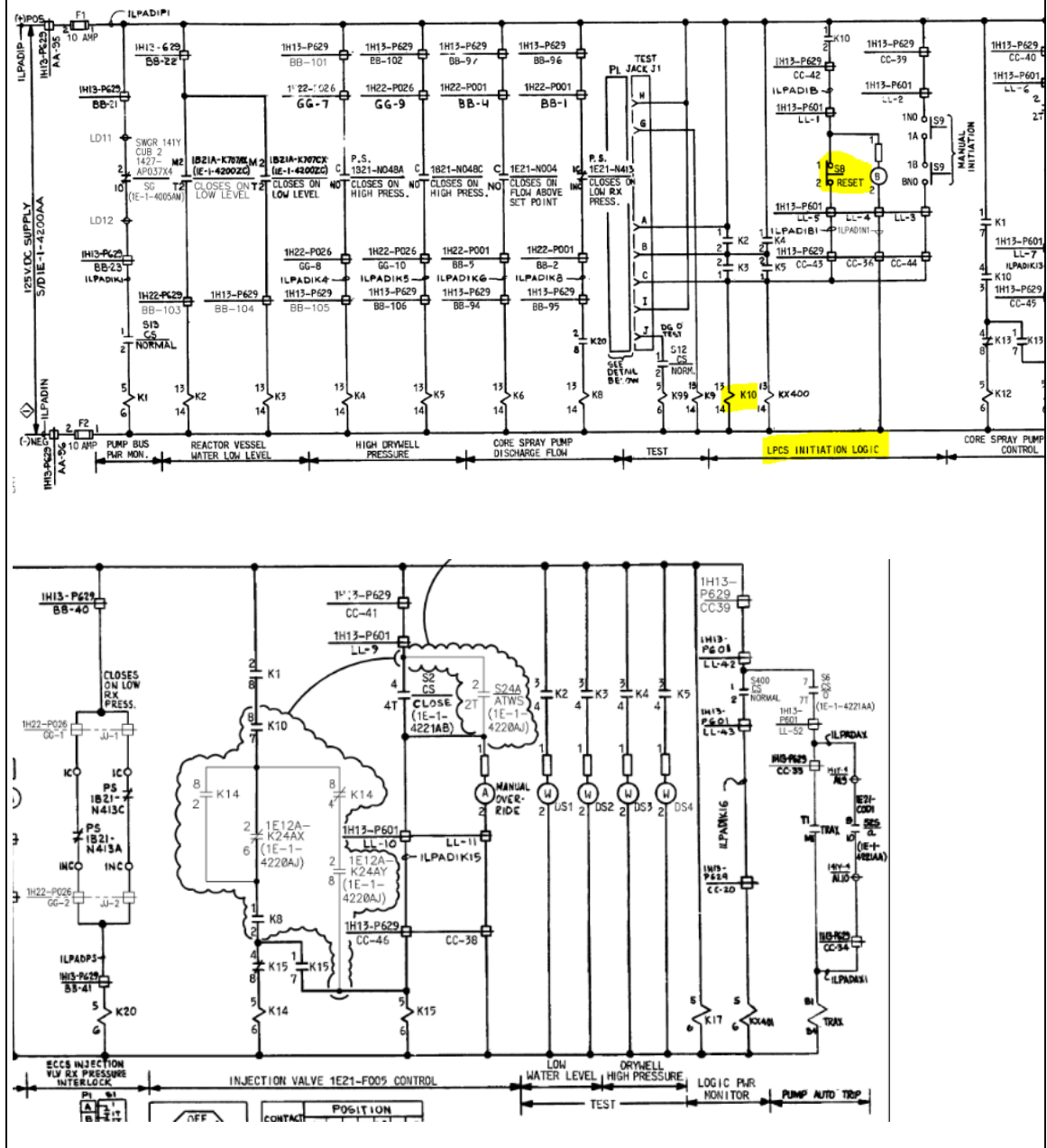
LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: “(1) 1H13-P601; (2) reset the system initiation logic “is the correct answer.

The pushbutton is located on the 1H13-P601, The low RPV level, high Drywell pressure, and manual LPCS initiation signals seal in once received. The RHR A/LPCS Initiation Reset Pushbutton resets a Division I initiation signal, but will not cause components to stop or realign automatically if running from the initiation signal.

Per Dwg 1E-1-4221AD

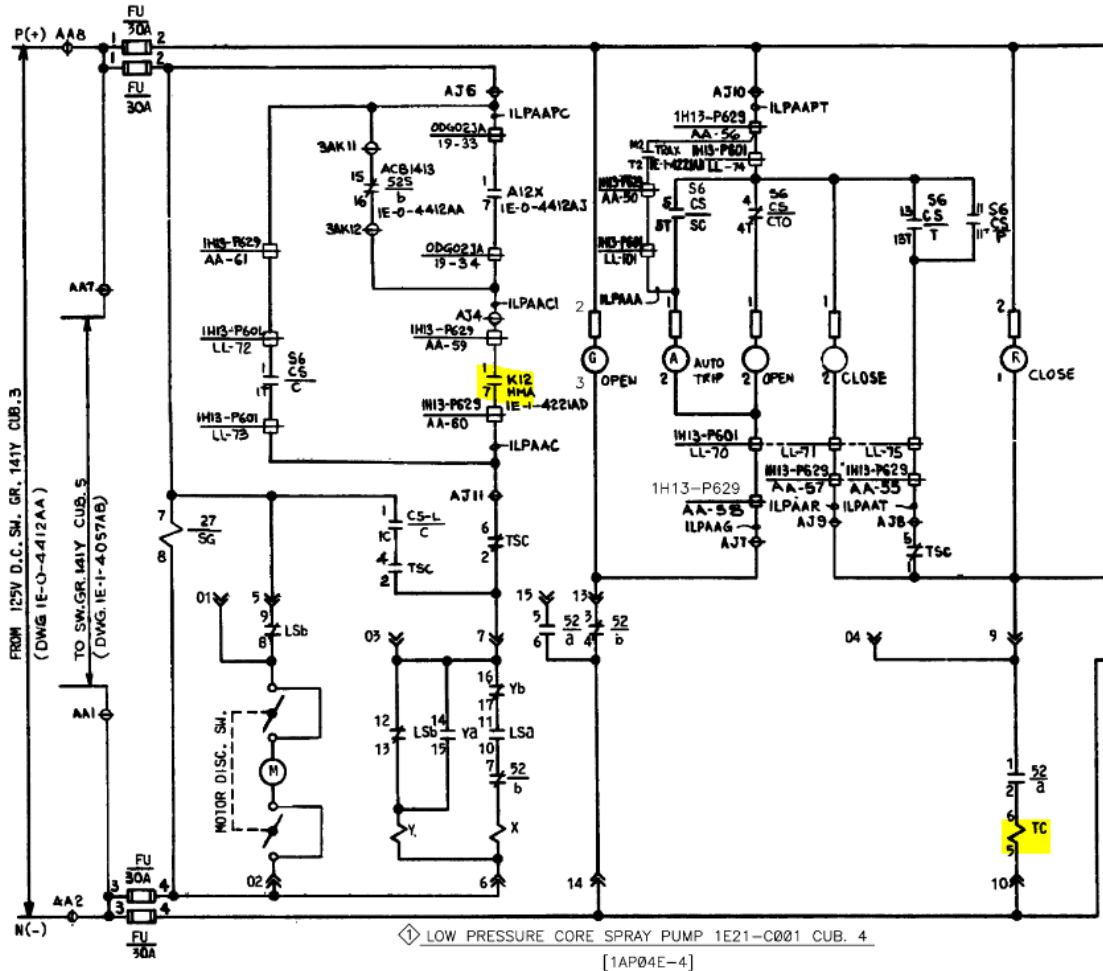


Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Per Dwg 1E-1-4221AA



Distractor 1 is plausible because the P603 panel is one of the MCR panels, RCIC controls are powered from Div 1 DC, are adjacent to the LPCS pump controls and RCIC auto initiates on low level, but incorrect because RCIC is not an ECCS system and RCIC initiation logic is not tied to the Div 1 ECCS initiation signal.

Distractor 2 is plausible because the P603 panel is one of the MCR panels and some initiation logics will result in shutdown of the started equipment, but incorrect because the reset button is located on the P601 panel and once the LPCS pump is started, only manipulation of the control switch or a loss of power will shut down the pump.

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Distractor 3 is plausible because reset of some initiation logics will result in valve closure, but incorrect because once the injection valve is opened, only manipulation of the control switch will close the valve.

Reference: Dwg 1E-1-4221AA, Dwg 1E-1-4221AB, Dwg 1E-1-4221AD

Reference provided during examination: None

Cognitive level: Memory

10CFR Part 55: 41.7

Importance Level (RO/SRO): 3.6

Tier: 2 Group: 1

PRA: No

K/A: 209001 A4.15 - Ability to manually operate and/or monitor the (SF2, SF4 LPCS) LOW PRESSURE CORE SPRAY SYSTEM in the control room: Initiation reset

Safety Function: 2 – Inventory control

SRO Justification: N/A

Question Source: Bank

Question History: None

Comments:

Objective #: 063.00.07

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

31

ID: 2544277

Points: 1.00

HPCS provides high pressure spray ____ (1) ____ to provide inventory control and spray cooling for ____ (2) ____ during large breaks.

- A. (1) below the reactor core
(2) the core shroud
- B. (1) below the reactor core
(2) partially covered fuel assemblies
- C. (1) above the reactor core
(2) the core shroud
- D. (1) above the reactor core
(2) partially covered fuel assemblies

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "(1) above the reactor core (2) partially covered fuel assemblies" is the correct answer.

HPCS spray nozzles are spaced in the RPV around the spargers to spray the water radially over the core and into the fuel assemblies. For large breaks the HPCS system cools the core by a spray.

Per UFSAR 6.3.2.2.1

The principal HPCS equipment is located outside the primary containment. Suction piping is provided from the suppression pool. The suppression pool water source assures a closed cooling water supply for extended operation of the HPCS system. After the HPCS injection piping enters the vessel, it divides and enters the shroud at two points near the top of the shroud. A semicircular sparger is attached to each outlet. Nozzles are spaced around the spargers to spray the water radially over the core and into the fuel assemblies. The HPCS injection piping is provided with an isolation valve on each side of the containment barrier. Remote controls for operating the valves and diesel generator are provided in the plant control room. The controls and instrumentation of the HPCS system are described, illustrated, and evaluated in detail in Chapter 7.0.

The HPCS system is designed to cool the reactor core sufficiently to prevent fuel cladding temperatures from exceeding the 10 CFR 50 limit of 2200° F following an break in the nuclear system piping. The system is designed to pump water into the reactor vessel over a wide range of pressures. For small breaks that do not result in rapid reactor depressurization, the system maintains reactor water level and depressurizes the vessel. For large breaks the HPCS system cools the core by a spray.

Distractor 1 is plausible because HPCS is designed to spray the reactor core, but incorrect because the spray spargers are located above the reactor core

Distractor 2 is plausible because HPCS is designed to spray the reactor core to cool partially covered fuel assemblies, but incorrect because HPCS injects above the core.

Distractor 3 is plausible because HPCS sprays above the top of fuel, but incorrect because cooling the core shroud is not the design of HPCS.

Reference: UFSAR 6.3.2.2.1

Reference provided during examination: None

Cognitive level: High

10CFR Part 55: 41.7

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Importance Level (RO/SRO): 3.7

Tier: 2 Group: 1

PRA: No

K/A: 209002 K1.12 - Knowledge of the physical connections and/or cause and effect relationships between the (SF2, SF4 HPCS) HIGH PRESSURE CORE SPRAY SYSTEM and the following systems: Reactor vessel and internals

Safety Function: 4 – Heat removal from reactor core

SRO Justification: N/A

Question Source: New

Question History: N/A

Comments:

Objective #: 061.00.04

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

32

ID: 2504715

Points: 1.00

Unit 1 was at 100% RTP when an ATWS occurred.

Actions are being taken IAW LGA-010, ATWS RPV CONTROL (1-3).

- The keylock switch for 1A Standby Liquid Control (SBLC) was placed in SYS A
- 1A Squib valve status light is lit
- 1B Squib valve status light is lit
- Annunciator 1H13-P603-A105, SBLC Squib Valve Continuity Loss, is NOT in alarm
- SBLC Pump Pressure is 1400 psig and steady

The squib valves (1) functioned correctly.

(2) What is the NEXT required action?

- A. (1) have
(2) Swap SBLC injection to SYS B
- B. (1) have
(2) Continue injecting with SYS A until SBLC tank level is 100 gallons
- C. (1) have not
(2) Swap SBLC injection to SYS B
- D. (1) have not
(2) Continue injecting with SYS A until SBLC tank level is 100 gallons

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: “(1) have not (2) Swap SBLC injection to SYS B” is the correct answer.

The squib valves did not fire as indicated by lit status lights and the failure of 1H13-P603-A105 to alarm.

Per LES-SC-101, Both squib valves will fire when either Sys A or Sys B is selected for either pump. The Squib valve status lights are normally lit indicating continuity through the firing circuit. The status lights will be OFF/ NOT lit after the squib valves are fired. For this case the A squib valve did not fire when the system was initiated. The Alarm monitors both squib valves and will come in when either squib valve loses continuity.

Per LGA-SC-101 if the initiation sequence for SBLC does not occur properly then an attempt is made to transfer to the other injection system by placing the keylock switch to off for the started system and placing the other system's keylock switch to the SYS B position in this case.

Distractor 1 is plausible because some indicating lights illuminate to show actuation, meaning the squib valves worked properly, and the SBLC pump discharge pressure is at the relief valve pressure indicating blockage in the system which would require swapping to SYS B, but incorrect because both valves should have fired causing both light indications to extinguish.

Distractor 2 is plausible because when SBLC is injecting properly the next required action will be waiting until SBLC tank level is 100 gallons, but incorrect because both valves should have fired causing both light indications to extinguish.

Distractor 3 is plausible because the required next step with SBLC injecting is to continue injecting until SBLC tank level is 100 gallons, but incorrect because SBLC is not injecting due to the failure of the squib valves to fire.

Reference: LES-SC-101 Rev 19; LOR-1H13-P603-A105, Rev 4; Dwg M-99 rev AD, LGA-SC-101 rev 5; LGA-010 rev 20

Reference provided during examination: None

Cognitive level: High
10CFR Part 55: 41.5

Importance Level (RO/SRO): 4.1/4.0
Tier: 2 Group: 1

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

PRA: No

K/A: 211000 A2.02 - 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: Failure of squib valves to open (CFR: 41.5)

Safety Function: 1

SRO Justification: N/A

Question Source: New

Question History:

Comments:

Objective #: 028.00.11 - EVALUATE system indications/responses and determine if indications/responses are expected and normal

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

33

ID: 2504847

Points: 1.00

Unit 1 is at 100% RTP.

- SBLC Tank is 4800 gallons
- SBLC tank concentration is 13%
- SBLC tank temperature is 60°F

Is LCO entry required? Why or why not?

- A. Yes; Concentration is too LOW for the given Volume
- B. Yes; Concentration is too HIGH for the given Volume
- C. Yes; Temperature is too LOW for the given Concentration
- D. No LCO entry required; parameters plot in the "Acceptable Operation Region"

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Yes; Temperature is too LOW for the given Concentration" is the correct answer.

Per Tech Spec Tables 3.1.7-1 4800 gallons and 13% concentration falls in the acceptable region and per 3.1.7-2 13% concentration at 60 degrees falls on the Unacceptable side of the curve which requires both trains of SBLC to be declared inoperable per TS 3.1.7.

Distractor 1 is plausible because volume versus concentration could plot in the unacceptable region, but incorrect because the minimum concentration for 4800 gallons is 12.53%.

Distractor 2 is plausible because volume versus concentration plots inside the shaded region which could be unacceptable, but incorrect because the shaded region of figure 3.1.7-1 is the acceptable operation region.

Distractor 3 is plausible because volume versus concentration plots within the shaded area and concentration versus temperature plots below the line, but incorrect because the acceptable operation region of figure 3.1.7-2 is above the line.

Reference: TS 3.1.7

Reference Provided during the examination: LOS-AA-S101 Figure A-4 and A-5 with the words "Administrative Acceptable Region", "T.S. 3.1.7 Acceptable Operation Region", "Acceptable Operation" and "Unacceptable Operation" removed

Cognitive level: High
10CFR Part 55: 41.7

Level (RO/SRO): 3.9 / 4.6
Tier: 2 Group: 1

PRA: No
K/A: 211000 - G2.2.42 - STANDBY LIQUID CONTROL SYSTEM /
EQUIPMENT CONTROL - Ability to recognize system parameters that are entry-level conditions for technical specifications

Safety Function: 1
SRO Justification: N/A

Question Source: New
Question History:

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Comments:
Objective #028.00.17

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

34

ID: 2504848

Points: 1.00

Unit 2 is at 100% RTP.

2B RPS MG set trips.

What PCIS valves will close?

- A. Outboard Main Steam Isolation Valves (Group 1)
- B. RBCCW outboard supply to the drywell (Group 2)
- C. Reactor Core Isolation Cooling (RCIC) outboard steam supply (Group 9)
- D. Instrument Nitrogen regulated supply to the drywell (Group 10)

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Instrument Nitrogen regulated supply to the drywell (Group 10)" is correct.

Loss of B RPS will result in PCIS Isolations of groups 1, 2, 3, 5, 6, 7, & 10 (except MSIVs. VP and WR).

Per LOA-RP-201

B.2 Loss Of Power To RPS Bus B

NOTE

On a loss of the B RPS bus, the following occurs:

- APRMs B, D, and F lose power.
- Channels B, D, F and H Oscillation Power Range Monitoring (OPRM) Systems will lose power.
- Radiation Monitors for Channels B and D Main Steam Line, Reactor Building Vent, and Fuel Pool Vent lose power.
- The IN System is lost due to a PCIS Group 10 Isolation.
- 2IN017, DW Pneumatic Supply Valve, closes.

15. CHECK Inboard and Outboard
PCIS valves have closed for
Groups 1-3, 5-7, and 10 except
for:

- MSIVs
- PCCW
- RBCCW
- 1B33-F020

Distractor 1 is plausible because RPS B powered solenoids have lost power, but incorrect because PCIS group 1 solenoids for the MSIVs are powered from A and B RPS to prevent MSIV closure upon loss of power to a single RPS bus.

Distractor 2 is plausible because a group 2 isolation signal is received, but incorrect because the RBCCW valves will not reposition on this loss of power.

Distractor 3 is plausible because RCIC is a PCIS isolation group, but incorrect because PCIS group 9 is not initiated by this loss.

Reference: LOA-RP-201 UNIT 2 LOSS OF REACTOR PROTECTION SYSTEM POWER rev 18 ; LOA-PC-201 PRIMARY/SECONDARY CONTAINMENT TROUBLE rev 24; 1E-2-4203AA rev P, and 1E-2-4203AF rev S
Reference provided during examination: None

Cognitive level: Memory

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

10CFR Part 55: CFR: 41.7 / 45.4

Level (RO/SRO): 3.9

Tier: 2 Group: 1

PRA: No

212000 K3.02 -Knowledge of the effect that a loss or malfunction of the (SF7 RPS) REACTOR PROTECTION SYSTEM will have on the following systems or system parameters: PCIS/NSSSS

Safety Function: 7

SRO Justification: N/A

Question Source: New

Question History:

Comments:

Objective # 049.00.14

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

35

ID: 2504851

Points: 1.00

Unit 1 is in STARTUP.

IRMs are on Range 1.

Which Intermediate Range Monitor (IRM) Rod Block is bypassed?

- A. IRM High
- B. IRM Inoperable
- C. IRM Downscale
- D. IRM Detector Not Full In

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "IRM Downscale" is the correct answer.

Per LOP-NR-02 when the IRM range switch is in Range 1 and the mode switch is not in RUN the downscale trip is bypassed.

D. LIMITATIONS

- D.1 The following signals will initiate a rod block when the reactor mode switch is NOT in RUN.
- D.1.1 Downscale trip, bypassed when IRM is in Range 1.
- D.1.2 Upscale trip.
- D.1.3 Any one IRM detector NOT fully inserted into the core.
- D.1.4 IRM Inoperative.

IRM High, IRM Downscale, IRM INOP, and IRM Detector Wrong Position/ IRM not full in are all bypassed in RUN but ONLY IRM downscale is bypassed by the range selector switch being in RANGE 1.

Distractor 1 is plausible because it initiates an IRM rod block and is automatically bypassed by a switch position, but incorrect because only the downscale trip is bypassed in Range 1.

Distractor 2 is plausible because it initiates an IRM rod block and is automatically bypassed by a switch position, but incorrect because only the downscale trip is bypassed in Range 1.

Distractor 3 is plausible because it initiates an IRM rod block and is automatically bypassed by a switch position, but incorrect because only the downscale trip is bypassed in Range 1.

Reference: LOP-NR-02, Rev 21; LP 042, Rev 9 IRMs

Reference provided during examination: None

Cognitive level: Memory

10CFR Part 55: 41.7

Level (RO/SRO): 4.0

Tier: 2 Group: 1

PRA: No

215003 K4.01 - Knowledge of (SF7 IRM) INTERMEDIATE RANGE MONITOR SYSTEM design features and/or interlocks that provide for the following: Rod withdrawal blocks

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Safety Function: 7- Instrumentation

SRO Justification: N/A

Question Source: Bank

Question History:

Comments:

Objective #042.00.08

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

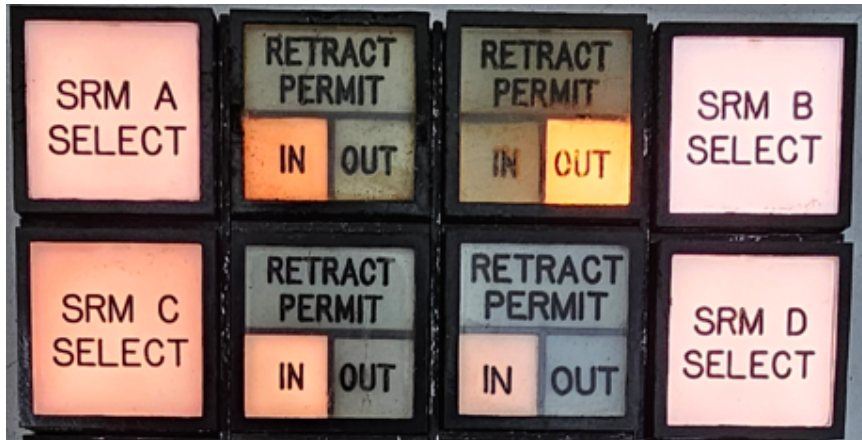
36

ID: 2504854

Points: 1.00

Unit 1 is in STARTUP preparing for rod withdrawals.

The Drive In/Driving In Light is LIT.



- (1) Without addressing the indicated issue, is rod withdrawal allowed?
(2) What action is required?

- A. (1) No
(2) Bypass SRM B
- B. (1) Yes
(2) Bypass SRM B
- C. (1) No
(2) Place SRM B function switch to STANDBY
- D. (1) Yes
(2) Place SRM B function switch to STANDBY

Answer: A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "(1) No (2) Bypass SRM B" is the correct answer.

The indication given is one of a stuck detector. Procedures require operators to stop all rod motion and bypass the detector that fails to move.

Per LOA-NR-101

B.1 SRM/IRM Insert/Withdraw Failure

- | | | | |
|----|---|-----|--|
| 1. | STOP all control rod motion/power changes | | |
| 2. | VERIFY Detector Drive and Position Indication POWER ON light - ILLUMINATED. | 2.1 | REPLACE the bulb. |
| | | 2.2 | If POWER ON light is <u>NOT</u> lit, GO TO Step 4. |

NOTE

Some of the SRM and IRM drive motors may have been upgraded from 1/8 Hp to 1/4 Hp. The 1/4 Hp motors will already have 3 amp fuses. If these fuses blow, they may be replaced, but not with higher amperage fuses.

- | | | | |
|----|--------------------------------------|-----|---|
| 3. | VERIFY desired detectors - SELECTED. | | |
| 4. | CHECK selected detectors - MOVE. | 4.1 | BYPASS affected channel. |
| | | 4.2 | REPLACE fuses on Attachment A as necessary per CC-AA-206, Fuse Control Program. |

Distractor 1 is plausible because the correct operator action is to bypass SRM B, but incorrect because rod moves are not allowed.

Distractor 2 is plausible because rod moves are not allowed, but incorrect because the SRM will be bypassed to prevent inaccurate nuclear instrumentation inputs to protective systems.

Distractor 3 is plausible because the Standby function will produce a rod block that will prevent rod movement, but incorrect because rod moves are not allowed until SRM B is bypassed.

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Reference: LOA-NR-101 NEUTRON MONITORING TROUBLE rev 23, LOP-NR-01 SOURCE RANGE MONITOR (SRM) OPERATION rev 19
Reference provided during examination: Embedded picture of IRM/SRM drive control panel

Cognitive level: High
10 CFR Part 55 Content:41.5

Importance Level (RO/SRO): 3.1/3.2
Tier: 2 Group: 1

K/A: 215004 A2.03 - Ability to (a) predict the impacts of the following on the (SF7 SRMS) SOURCE RANGE MONITOR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Stuck detector

Safety Function: 7 - Instrumentation
SRO Justification: N/A

Question Source: New
Question History:

Comments:

Objective: 041.00.11

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

37

ID: 2544301

Points: 1.00

Unit 2 is at 100% RTP when 2A RPS MG set trips.

2A RPS bus has been transferred to the alternate power supply.

OPRMs ____ (1) ____ must be reset, because of a loss of the associated ____ (2) ____.

- A. (1) A, C, E
(2) APRM inputs
- B. (1) B, D, F
(2) APRM inputs
- C. (1) A, C, E, G
(2) OPRM power supply
- D. (1) B, D, F, H
(2) OPRM power supply

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "(1) A, C, E, G; (2) OPRM power supply" is the correct answer.

Per LOA-RP-201 B.1 Note:

On a loss of power to the A RPS bus, the following occurs:

- APRMs A, C, E lose power
- OPRMs A, C, E, G will lose power

Distractor 1 is plausible because the A, C, E OPRMs take their inputs from the A, C, E APRMs powered from RPS and if an OPRM is bypassed, bypassing the APRM associated with the 'companion' OPRM will result in an OPRM trouble alarm, but incorrect because A RPS bus powers A, C, E, and G OPRMs which will trip and there are no OPRMs in bypass.

Distractor 2 is plausible because the B, D, F OPRMs take their inputs from the B, D, F APRMs powered from RPS and if an OPRM is bypassed, bypassing the APRM associated with the 'companion' OPRM will result in an OPRM trouble alarm, but incorrect because A RPS powers A, C, E, and G OPRMs which will trip and there are no OPRMs in bypass.

Distractor 3 is plausible because the B, D, F, H OPRMs are powered from RPS, but incorrect because A RPS powers A, C, E, and G OPRMs which will trip.

Reference: LOA-RP-201, Rev 18; LOR-2H13-P603-B507, Rev 6; LOA-NR-201 Rev 24

Reference provided during examination: None

Cognitive level: Memory
10CFR Part 55: 41.7

Importance Level (RO/SRO): 3.6
Tier: 2 Group: 1

PRA: No
K/A: 215005 K2.03 - (SF7 PRMS) AVERAGE POWER RANGE
MONITOR/LOCAL POWER RANGE MONITOR Knowledge of electrical power supplies to the following: OPRM channels

Safety Function: 7- Instrumentation
SRO Justification: N/A

Question Source: New
Question History:

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Comments:

Objective #: 055.00.13

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

38

ID: 2505059

Points: 1.00

Unit 1 was at 100% RTP when a scram occurred.

- RCIC is the only injection source available and is being operated with the flow controller in Automatic with flow set at 600 gpm
- RPV water level is -15 inches and stable
- Reactor pressure is 900 psig and lowering 1 psig/min

Subsequently, RCIC flow input signal to the controller fails HIGH.

With NO operator action:

Turbine speed will ____ (1) ____ and reactor water level will ____ (2) ____.

- A. (1) rise
(2) rise
- B. (1) lower
(2) lower
- C. (1) NOT change
(2) rise
- D. (1) NOT change
(2) lower

Answer: B

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "(1) Lower, (2) Lower" is the correct answer.

UFSAR 5.4.6.3

The RCIC turbine has a speed governor that is positioned by the demand signal from the flow controller. Maximum output from the controller corresponds to maximum turbine speed.

Automatic flow control adjusts turbine speed as necessary to control flow at the setpoint. With RCIC Flow Feedback to the controller failing high, the controller will develop a signal to continuously lower turbine speed to compensate for the difference between the flow feedback and the flow setpoint. Since RCIC is the only injection source reactor water level will lower.

Distractor 1 is plausible because this is the response of the RCIC system if the controller flow feedback has failed low, but incorrect because flow feedback has failed high resulting in a slower turbine speed and lowering reactor level.

Distractor 2 is plausible because some controllers automatically transfer to Manual mode where turbine speed would be constant and changing reactor pressure would cause level to change, but incorrect because the RCIC controller will not auto transfer to manual mode and speed and level will lower.

Distractor 3 is plausible because some controllers automatically transfer to Manual mode where speed would remain constant and changing reactor pressure would cause level to change, but incorrect because the RCIC controller will not auto transfer to manual mode and speed will lower.

Reference: GEFES BWR / COMPONENTS / CHAPTER 8 / CONTROLLERS AND POSITIONERS, Rev 4b; UFSAR 5.4.6.3

Reference provided during examination: None

Cognitive level: High

10CFR Part 55: 41.7

Importance Level (RO/SRO): 4.2

Tier: 2 Group: 1

PRA: No

K/A: 217000 K3.01 - Knowledge of the effect that a loss or malfunction of the (SF2, SF4 RCIC) REACTOR CORE ISOLATION COOLING SYSTEM will have on the following systems or system parameters: Reactor water level

Safety Function: 2 - Reactor Water Inventory Control

SRO Justification: N/A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Question Source: New

Question History:

Comments:

Objective #: 032.00.16

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

39

ID: 2505061

Points: 1.00

Unit 1 was at 100% RTP when a LOCA occurred.

- RPV water level is -150 inches and lowering
- RPV pressure is 900 psig and steady
- DW pressure is 1.0 psig and steady
- Division 1 ECCS pumps started
- Division 2 ECCS pumps did not start automatically

What will cause ADS to initiate?

- A. LPCS pump is started ONLY
- B. 1B RHR pump is started ONLY
- C. 720 seconds have elapsed ONLY
- D. RPV Water Level 1 is reached ONLY

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "720 seconds have elapsed ." is the correct answer

Per LOP-MS-03 PREPARATION FOR STANDBY OPERATION OF THE AUTOMATIC DEPRESSURIZATION SYSTEM, Section D.6 lists the requirements for ADS auto initiation which are:

- High Drywell Pressure (1.77 psig) OR high drywell Pressure Bypass after 9 minutes (598 second TS required) of RPV Level 1
- RPV level 3 (11.0 inches)
- RPV level 1 (-147.0 inches)
- 105 second time delay (118 TS required)
- Discharge pressure on ECCS pump in division being initiated:
 - LPCS
 - RHR

Drywell pressure is not high enough for automatic initiation of ADS, so the 598 second time delay is required to bypass the lack of the high drywell pressure. 598 seconds + 118 seconds = 716 seconds. At 720 seconds ADS will initiate with no further actions taken

Distractor 1 is plausible because LPCS pump is required to initiate ADS, but incorrect because LPCS is a division 1 component and the drywell bypass timer still needs to expire.

Distractor 2 is plausible because 1B RHR pump is required to initiate the Division 2 ADS and most protective features require both divisions to actuate, but incorrect because only 1 division is required to initiate ADS.

Distractor 3 is plausible because RPV water level 1 is required for ADS initiation, but incorrect because RPV water level is below RPV water level 1 which is the requirement to automatically initiate ADS.

Reference: LOP-MS-03 PREPARATION FOR STANDBY OPERATION OF THE AUTOMATIC DEPRESSURIZATION SYSTEM rev 13, ITS 3.3.5.1.

Reference provided during examination: None

Cognitive level (Memory/High): High
10CFR Part 55:41(b)(7)

Level (RO/SRO): 4.2
Tier: 2 Group: 1

PRA: No

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

K/A: 218000 A4.02 Ability to manually operate and/or monitor in the control room: ADS logic initiation

Safety Function: 3

SRO Justification: N/A

Question Source: New

Question History:

Comments:

Objective #062.00.07

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

40

ID: 2505193

Points: 1.00

Unit 1 is at 15% RTP.

- Unit 1 Turbine Generator is connected to the grid
- Main Condenser backpressure is 27" Hg and degrading

Assuming NO operator action, the reactor will automatically scram on...

- A. Turbine Stop Valve (TSV) closure.
- B. Turbine Control Valve (TCV) closure.
- C. Turbine Bypass Valve (TBV) closure.
- D. Main Steam Isolation Valve (MSIV) closure.

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Main Steam Isolation Valve (MSIV) closure" is the correct answer. When the MSIVs close on low vacuum, the reactor will scram.
Per LOP-CX-06 PRIMARY CONTAINMENT ISOLATION STATUS DISPLAY:
Group 1 isolation occurs on Low Condenser Vacuum of 3.8" Hg vacuum

PCIS ISOLATION SIGNALS

<u>Group I Isolation</u>
Level 1 (-137")
High Main Steam Line Flow (134%)
Low MSL Pressure with Mode Switch in RUN (826.5 psig)
Low Condenser Vacuum (3.8" Hg vacuum)
Main Steam Tunnel Ventilation High Differential Temperature (66.4°F)

Per LOR-1H13-P603-B105 and -B109: a Reactor automatic scram will occur with MSIVs less than 92% (86.3% TS required) open with the mode switch in RUN.

With reactor power at 15% and the main turbine connected to the electrical grid, Unit 1 mode switch is in RUN with the MSIVs open. with lowering vacuum a group 1 isolation will occur at 3.8" Hg condenser vacuum closing all MSIVs and causing an automatic reactor scram.

Distractor 1 is plausible because the turbine stop valves will close when the main turbine trips on low vacuum, about 22" Hg vacuum, but incorrect because Reactor Scram on a turbine trip is automatically bypassed when reactor power is less than 25% power.

Distractor 2 is plausible because the turbine control valves will fast close when the main turbine trips on low vacuum, about 22" Hg vacuum, but incorrect because Reactor Scram on a turbine trip is automatically bypassed when reactor power is less than 25% power.

Distractor 3 is plausible because the turbine bypass valves will close on low vacuum at about 7" Hg vacuum and closure of the turbine bypass valves will result in an RPV pressure rise and eventual scram on high RPV pressure, but incorrect because the turbine bypass valves will close at the same condenser backpressure as the MSIVs and the MSIVs will cause a direct reactor scram when they begin to stroke closed.

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Reference: LOP-CX-06 PRIMARY CONTAINMENT ISOLATION STATUS
DISPLAY, Rev 8; LOR-1H13-P603-B105/109, Rev 4, LOR-1PM03J-B511 rev 7
Reference provided during examination: None

Cognitive level: High
10CFR Part 55: 41.5

Level (RO/SRO): 3.1
Tier: 2 Group: 2

PRA: No
223002 A1.12 - Ability to predict and/or monitor changes in parameters
associated with operation of the (SF5 PCIS/NSSS) PRIMARY CONTAINMENT
ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUTOFF, including:
Condenser vacuum

Safety Function: 5
SRO Justification: N/A

Question Source: Bank
Question History:

Comments:
Objective #: 070.00.16

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

41

ID: 2505196

Points: 1.00

Unit 1 is at 100% RTP.

Reactor pressure rises to 1100 psig before lowering to 880 psig.

What is the total number of safety relief valves that have opened and are now closed?

A. 2

B. 8

C. 11

D. 13

Answer: B

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "8 " is the correct answer.

Per LOA-SRV-101: 2 SRVs will open at 1076 psig, 4 SRVs will open at 1086 psig, and 2 SRVs will open at 1096 psig for a total of 8 SRVs.

SRV PRESSURE SETPOINTS

NOTE

If EC 366749 is installed, the Low Low Set function of the S and U SRVs will be transferred to the V and E SRVs respectively. The parenthetical open and closed setpoints for V and E SRVs will be active if EC 366749 is installed.

SRV	Relief Valve OPEN	Relief Valve CLOSE	Low-Low Set OPEN	Low-Low Set Re-open	Low-Low Set CLOSE	Safety Valve OPEN	Safety Valve CLOSE
C	1086	986	1086	--	946	1175	1140
D	1086	986	1086	--	946	1175	1140
E	1086	986	1086 (1076)	-- (1006)	946 (896)	1175	1140
F	1096	996	--	--	--	1185	1149
H	1106	1006	--	--	--	1195	1159
K	1086	986	1086	--	946	1175	1140
L	1106	1006	--	--	--	1195	1159
M	1106	1006	--	--	--	1195	1159
P	1096	996	1096	--	946	1185	1149
R	1116	1016	--	--	--	1205	1169
S	1076	976	1076	1046	926	1150	1115
U	1076	976	1076	1006	896	1150	1115
V	1116	1016	(1076)	(1046)	(926)	1205	1169

Distractors 1 is plausible because the first group of SRVs to open contains 2 SRVs, but incorrect because the relief setpoint has been exceeded for 8 SRVs.

Distractor 2 is plausible because the next setpoint will open a total of 11 SRVs, but incorrect because pressure only reached 1100 psig and only 8 SRVs would have opened.

Distractor 3 is plausible because the pressure is above the close setpoint of all 13 SRVs, but incorrect because the close set point is when the valves will reseal if the open setpoint has been exceeded.

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Reference: LOA-SRV-101 UNIT 1 STUCK OPEN SAFETY RELIEF VALVE rev 9; LOR-1H13-P603-B107 DIV 2 ARI LOGIC INITIATED rev 4; System Description 070 Main Steam, Rev 12
Reference provided during examination: None

Cognitive level (Memory/High): Memory
10CFR Part 55: 41.5

Importance Level (RO/SRO): 4.3
Tier: 2 Group: 1

PRA: No
K/A: 239002 A1.04 - Ability to predict and/or monitor changes in parameters associated with operation of the (SF3 SRV) SAFETY RELIEF VALVES including: Reactor pressure

Safety Function: 3
SRO Justification: N/A

Question Source: Bank
Question History:

Comments:
Objective #070.00.07

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

42

ID: 2544541

Points: 1.00

Unit 2 was at 100% RTP when a LOCA occurred.

- All ECCS pumps are running
- Automatic Depressurization (ADS) was manually initiated
- SRVs C, D, E, S, and U indicate open
- All other SRVs remain closed

SRVs _____ also should have opened.

- A. E and L
- B. H and M
- C. K and P
- D. R and V

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Answer Explanation: "(1) R and V; (2) MCR P601 Panel" is the correct answer.

Per Drawings 1E-2-4201AA, AF, AG, the ADS SRV's are C, D, E, R, S, U, V (SCURVED). SRVs R and V should have opened with an ADS signal.

Distractor 1 is plausible because SRV E is one of the ADS valves, but incorrect because SRV L is not an ADS valve.

Distractor 2 is plausible because SRVs H and M will open under the same plant conditions, but incorrect because SRVs H and M open at the same pressure set point and are not part of the ADS SRVs .

Distractor 3 is plausible because SRV's K and P are part of one of the groupings of SRVs, but incorrect because the SRV Grouping C, D, E, K, P, S, U (DESK CUP) are Lo Lo set SRV's where K, P are not part of the ADS logic.

Reference: Drawings 1E-2-4201AA thru AK

Reference provided during examination: None

Cognitive level: High

10CFR Part 55: 41.7

Importance Level (RO/SRO): 4.1

Tier: 2 Group: 1

PRA: No

K/A: 239002 A3.01 - Ability to monitor automatic operation of the (SF3 SRV) SAFETY RELIEF VALVES including: SRV operation after ADS actuation

Safety Function: 3

SRO Justification: N/A

Question Source: New

Question History:

Comments:

Objective #: 070.00.11

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

43

ID: 2546668

Points: 1.00

Unit 1 is at 100% RTP.

- '1A' TDRFP trips
- Annunciator 1H13-P603-A409, Feedwater Control Reactor Water Low-Level 4, alarms

(1) How will Reactor Water Level Control system (RWLC) cause the Reactor Recirculation system to respond?

(2) What is the reason for the response?

- A. (1) RR pump Downshift;
(2) To prevent cavitation in RR FCVs
- B. (1) RR Flow Control Valve Runback;
(2) To prevent cavitation in RR FCVs
- C. (1) RR pump Downshift;
(2) Reactor power is beyond the capacity of the operating reactor feed pumps
- D. (1) RR Flow Control Valve Runback;
(2) Reactor power is beyond the capacity of the operating reactor feed pumps

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "(1) RR Flow Control Valve Runback; (2) Reactor power is beyond the capacity of the operating reactor feed pumps" is the correct answer.

An RR Low FW Capacity Runback is initiated when reactor level is <31.5-inches and inadequate FW system capacity exists as indicated by:

- 1) Total steam flow
- 2) both High and Low-pressure steam stop valves are not full-open, AND either of the following:
 - Hydraulic trip oil pressure is low, OR
 - the turbine trip device is in the trip position

The MDRFP status is determined by the position of its circuit breaker.

The FCV runback is driven by the RWLCS to reduce recirculation flow to within the power capabilities of the remaining reactor feed pumps. 1B TDRFP in this case until the MDRFP starts and starts feeding the RPV.

Per LOP-RL-01 OPERATION OF THE REACTOR LEVEL CONTROL SYSTEM.

The RR FCV Runback characteristics:

- o The RR FCV Runback is initiated when total steam flow exceeds feedwater system capacity and level drops below level 4.
- o The RR FCV Runback is terminated when level is increased above the level 4 setpoint when total steam flow is reduced 5% below the feedwater system capacity or when the RR FCV valve reaches minimum position. If a valid total steam flow signal is not available, the runback is terminated when level is increased above the level 4 setpoint when the RR FCV reaches minimum.
- o Assumed capacity for a TDRFP is 67%, and for the MDRFP is 33%.
- o The RR Runback feature does not require resetting.

Distractor 1 is plausible because the RR pump downshift occurs to prevent cavitation in the FCVs, but incorrect because the RR pump downshift does not occur until feedwater flow is sustained below 20% and with one TDRFP operating the feed flow capacity is 67%.

Distractor 2 is plausible because RR flow control valves will runback, but incorrect because the purpose of the FCV runback is to reduce the steaming rate to within the capacity of the remaining feedwater pumps.

Distractor 3 is plausible because the RR pump downshift is driven from reduced feedwater flow and the feedwater capacity is beyond the current reactor power,

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

but incorrect because the RR pump downshift does not occur until feedwater flow is sustained below 20% and with one TDRFP operating the feed flow capacity is 67%.

Reference: LOP-RL-01 OPERATION OF THE REACTOR LEVEL CONTROL SYSTEM rev 28; UFSAR Appendix G, UFSAR 7.7

Reference provided during examination: None

Cognitive level: High

10CFR Part 55: 41(b)(5)

Importance Level (RO/SRO): 3.8

Tier: 2 Group: 2

PRA: No

K/A: 259002 K5.02 Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Reactor Water Level Control System: Controller Operation

Safety Function: 2

SRO Justification: N/A

Question Source: New

Question History:

Comments:

Objective #:023.00.08

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

44

ID: 2505202

Points: 1.00

Unit 1 is at 100% RTP.

The purpose of the Standby Gas Treatment System (VG) Cooling Fan 1VG02C is to cool the...

- A. charcoal bed and remove the heat generated by the decay of fission products that became trapped within the VG filter train.
- B. primary fan and remove the heat generated by I^2R losses to prevent motor damage.
- C. processed air flow prior to entering the VG train to limit the air temperature to 150°F.
- D. electric heaters during operation to prevent carbon bed combustion.

Answer: A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Cools the VG Charcoal Bed and removes the heat generated by the decay of fission products that became trapped within the VG filter train." is correct.

The Cooling Fan 1(2)VG02C dissipates decay heat from fission products deposited on the charcoal filters. The fan takes suction from the Reactor Building atmosphere and discharges through the train to the exhaust stack.

Per UFSAR 6.5.1.1.1

- j. A standby cooling air fan is provided for each equipment train to remove heat generated by fission product decay on the HEPA filters and charcoal adsorbers after shutdown of the train.

The standby cooling air fan is conservatively sized to remove approximately 7700 Btu/hr of heat (generated by instantaneous deposition of iodine, on a HEPA filter bank and charcoal adsorbers) with less than a 50° F rise in cooling air temperature. This will limit the air temperature in the SGTS to 200° F maximum to prevent possible desorption and fire. Charcoal desorption temperature is given in ORNL-NSIC-65. No credit is taken for equipment or environment heat sink. Reactor building cooling air is routed through the shutdown train and exhausted to the atmosphere via the plant vent stack.

Distractor 1 is plausible because other motors have integral fans for cooling, but incorrect because the Cooling fan is interlocked to start when the Primary Fan has stopped and provide cooling to the train not the primary fan.

Distractor 2 is plausible because there is a design temperature limit for the train, but incorrect because the inlet air is assumed to be a maximum of 150°F and the Cooling Fan capacity limits air temperature to 200°F.

Distractor 3 is plausible because electric heaters are used to heat the air in the train to keep the Charcoal dry, but incorrect because the Cooling Fan is used after the train is shutdown and when the Electric Heater and Primary Fan are not operating.

Reference: UFSAR 6.5.1.1.1; LP 095 VG, Rev 13

Reference provided during examination: None

Cognitive level: Memory

10CFR Part 55: CFR: 41.5

Level (RO/SRO): 3.0

Tier: 2 Group: 1

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

PRA: No

K/A: 261000 K5.01 - Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF9 SGTS) STANDBY GAS TREATMENT SYSTEM: Heat removal mechanisms

Safety Function: 9

SRO Justification: N/A

Question Source: Bank

Question History:

Comments:

Objective # 095.00.04

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

45

ID: 2505224

Points: 1.00

Which of the following is a Unit 2, 4160V, Division I, ESF bus?

- A. 241Y
- B. 142X
- C. 232Y
- D. 241X

Answer: A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Answer Explanation: "241Y" is the correct answer.

Per Drawing 1E-2-4000AK, Bus 241Y supplies power to the Division 1 ESF loads.

Distractor 1 is plausible because 142X is a 4160V bus, but incorrect because it is a Unit 1 power supply and does not support ESF loads.

Distractor 2 is plausible because 232Y is an electrical bus and most ESF related buses have a 'Y' ending such as 235Y, but incorrect because it is a 480V bus and does not support ESF loads.

Distractor 3 is plausible because 241X is a Unit 2 4160V bus that can be tied to the Div 1 Diesel Generator with jumpers in place, but incorrect because it does not support ESF loads.

Reference: Drawing 1E-2-4000AK; LP 005 AC Distribution, Rev 20

Reference provided during examination: None

Cognitive level: Memory

10CFR Part 55: 41.7

Importance Level (RO/SRO): 3.8

Tier: 2 Group: 1

PRA: Yes

K/A: 262001 K2.03 - (SF6 AC) AC ELECTRICAL DISTRIBUTION Knowledge of electrical power supplies to the following: Major motor control centers/buses (480 volts and higher)

Safety Function: 6

SRO Justification: N/A

Question Source: Bank

Question History:

Comments:

Objective # 005.00.04

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

46

ID: 2505226

Points: 1.00

Unit 1 is at 100% RTP.

- Bus 112Y loses power
- Subsequently the Unit 1 Station Auxiliary Transformer (SAT) trips due to a failed sudden pressure relay

What is the status of the 1A DG and Bus 142Y?

- (1) 1A Diesel Generator
- (2) 4.16 kVAC Bus 142Y

- A. (1) Running
(2) Energized
- B. (1) Running
(2) De-energized
- C. (1) Off
(2) Energized
- D. (1) Off
(2) De-energized

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: “ (1) Off (2) de-energized “ is the correct answer.

With a loss of 112Y there is a loss of power to the loss of power relays for breaker 1421. Therefore, bus 142Y will not automatically swap to the UAT supplying with 112Y dead.

The Diesel Generator will not start on under voltage in this case (1E-1-4005AT). The logic for Switchgear 142Y under voltage relays are power from 112Y. So AP040X1 and AP040X2 relays never energize (even though an under-voltage condition is present). The 1427-AP040X3 relay never energizes. Therefore, the 1A D/G does NOT get a start signal (1E-1-4009AE).

Distractors 1 is plausible because safety related buses will fast transfer on a loss of normal power supply, but incorrect because with a loss of 112Y there is a loss of power to the relays that provide the loss of power monitoring.

Distractor 2 is plausible because there has been a loss of power to the control relays so an automatic trip of the breakers has not occurred and the 52/b relay prevents closing the 1A DG output breaker to energize the bus, but incorrect because with a loss of the relay power the 1A DG will not receive a start signal.

Distractor 3 is plausible because the 1A DG will not receive a start signal due to a loss of the relay power, but incorrect because a loss of the SAT will also result in de-energization to Bus 142Y.

Reference: Electrical Drawing 1E-1-4009AE rev T, 1E-1-4000AM rev E, 1E-1-4005AT rev P, 1E-1-4005AQ rev O, 1E-1-4005AS rev N, 1E-1-4009AA rev AB

Reference provided during examination: None

Cognitive level: High
10CFR Part 55:41.5

Importance Level (RO/SRO): 3.5
Tier: 2 Group: 1

PRA: Yes
K/A: 262001 K5.02 - Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF6 AC) AC ELECTRICAL DISTRIBUTION: Breaker control power

Safety Function: 6
SRO Justification: N/A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Question Source: Bank

Question History:

Comments:

Objective #:05.00.13

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

47

ID: 2544585

Points: 1.00

Unit 1 is at 100% RTP when annunciator 1PM01J-A111, Uninterruptible Power Supply Trouble, Alarms.

SER indicates:

- UPS Static Switch Alt Pos alarm
- UPS 480V Norm Sply Volt Lo alarm
- UPS Inverter Failure alarm

What is the status of the Plant Process Computer (PPC) Uninterruptible Power Supply (UPS)?

PPC UPS...

- A. lost alternate AC power
- B. transferred to DC power
- C. transferred to alternate AC power
- D. is no longer providing output power

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "transferred to alternate AC power" is the correct answer.

Loss of both the Normal AC Source and the Backup DC Source (or failure of the 25 KVA Inverter as indicated by the inverter failure alarm) will cause the Static Transfer Switch to swap to the Alternate AC Source.

Distractor 1 is plausible because the synchronization failure will alarm when either normal or alternate power is lost, but incorrect because the UPS 480V normal power supply alarm is in with the Static switch alternate position alarm indicating the UPS has transferred to the alternate power supply and the static switch output failure alarm is not in alarm indicating power is still being supplied by the PPC UPS.

Distractor 2 is plausible because the static inverter has a DC power supply and DC is the backup power supply to the static inverter if a failure occurs in the normal AC supply, but incorrect because the Inverter Failure alarm was receive indicating the static inverter is no longer supplying.

Distractor 3 is plausible because the inverter failure alarm is in which is associated with the Inverter failure light which will only be lit if output from the static inverter is lost, but incorrect because the alt switch output failure alarm did not occur.

Reference:

LOR-1PM01J-A111 rev 5; 1E-1-4007AA rev C; LOP-CX-08, Revision 12; 1E-1-4453AB rev D; 1E-1-4007AC rev D;
System Description 012, Figure 12-03, Rev 11

Reference provided during examination: None

Cognitive level: High
10CFR Part 55: 41.5

Importance Level (RO/SRO): 3.1
Tier: 2 Group: 1

PRA: No

K/A: 262002 A1.01 - Ability to predict and/or monitor changes in parameters associated with operation of the (SF6 UPS) UNINTERRUPTABLE POWER SUPPLY (AC/DC) including: Inverter outputs

Safety Function: 6
SRO Justification: N/A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Question Source: New

Question History:

Comments:

Objective #: 012.00.13

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

48

ID: 2544619

Points: 1.00

Unit 2 is at 100% RTP.

- Annunciator 2PM01J-B104, Div 2 125V DC Batt Charger trouble, alarms
- The 2BA Battery charger is damaged
- The Alternate battery charger is placed in service to the Div 2 125V DC bus

What are the expected conditions for the Div 2 125V DC bus and Battery?

Charger Voltage = 128 Vdc

Charger current = __ (1) __

Battery current = __ (2) __

- A. (1) 0 amps
(2) 0 amps
- B. (1) 0 amps
(2) 25 amps
- C. (1) 25 amps
(2) 0 amps
- D. (1) 25 amps
(2) 25 amps

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "(1) 25 amps; (2) 0 amps" is the correct answer.

The normal indications with the charger online are:

- Div 2 125 V Battery Volts indicate 127 Vdc
- Div 2 125 V Battery Amps indicate 0 amps
- Div 2 125 V Charger Volts indicate 127 Vdc
- Div 2 125 V Charger Amps indicate 25 amps

When the Charger fails, the load that the Charger was carrying will be transferred to the Battery as indicated by the 25 amp current flow from the battery. When the spare charger is placed in service, the charger will once again carry the load with small increased current output to restore the battery as well as carrying running DC loads. Battery current output will return to 0 amps.

Per LOP-DC-01

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

LOR-2PM01J-B104

Revision 4

August 22, 2005

Description: Div II 125VDC Batt Charger Trouble

Computer Print: Multiple (See Below)

Setpoint: Multiple (See below)

Sensor No: K1, 52, K3 2EY-DC047, 48 (TD Relay)

Alarm No: 2DC09A/B/C/D/E/F

Drawing: 1E-2-4440AA 1E-2-4008ZC, AH, AJ, AK

Activating Device: Multiple

**DIV 2 125V DC
BATT CHARGER
TROUBLE**

A. AUTOMATIC ACTIONS:

1. None.

B. OPERATOR ACTIONS:

1. DETERMINE cause for alarm condition, and CORRECT.
Possible causes are:
 - a. R0985 – Div 2 125V Charger Output Volt Lo/Fail
 - b. R0986 – Div 2 125V Charger AC Voltage Failure
 - c. R0987 – Div 2 125V Charger Feed Breaker Trip
 - d. R1026 – Div 2 125V Battery Discharge Hi
 - e. R1059 – Div 2 125V Batt Instrumentation Failure
 - f. R1060 – Div 2 125V Battery Charge Rate Hi
2. VERIFY proper operation of 125V Battery Charger per LOP-DC-01, Energizing, Startup, and Shutdown of a Battery Charger.
3. REFER to LOA-DC-201 Unit 2 DC Power System Failure.

Distractor 1 is plausible because if all the loads have been stripped from the bus and not restored, but incorrect because the loads will be maintained on the battery to prevent loss of breaker function and DC powered logic.

Distractor 2 is plausible because it is possible indication, but incorrect because the current flow from the battery only is indication that the charger is not carrying the system load.

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Distractor 3 is plausible because it is possible for both the charger and battery to carry load, but incorrect because a charger setup properly with carry all the system load and charge the battery also.

Reference: LOR-2PM01J-B104, Rev 4; LOA-DC-201, Rev 25; LOP-DC-01, Rev 50; LP 006 DC Distribution, Rev 12, pg. 4

Reference provided during examination: None

Cognitive level: Memory

10CFR Part 55: 41.7

Importance Level (RO/SRO): 3.2

Tier: 2 Group: 1

PRA: Yes

K/A: 263000 A4.06 - Ability to manually operate and/or monitor the (SF6 DC) DC ELECTRICAL DISTRIBUTION in the control room: Transfers

Safety Function: 6

SRO Justification: N/A

Question Source: New

Question History:

Comments:

Objective #: 006.00.11

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

49

ID: 2544675

Points: 1.00

Unit 2 is at 100% RTP with the Common DG running fully loaded for surveillance testing.

DC bus 211Y trips.

The Common DG will. . .

- A. continue to run fully loaded.
- B. trip due loss of DG Exciter field.
- C. shutdown due to a loss of the Fuel Priming pump.
- D. overheat and trip due to a loss of DC Soak Back pump.

Answer: A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "continue to run fully loaded." is the correct answer.

Per LOA-DC-201:

CAUTION: The following are NOT operable:

- 0 DG capability to auto start on Unit 2 ECCS actuation/Bus UV.
- 0 DG control power/protective trips from Unit 2.
- Bus 241Y breaker control power/protective trips.
- LPCS/LPCI injection flowpaths.

If the 0 DG was running loaded prior to the loss of 211Y operators are directed to locally unload and secure the DG.

Distractor 1 is plausible because DC power is required to flash the exciter field, but incorrect because once the field is flashed after DG start DC power is no longer required to maintain it.

Distractor 2 is plausible because fuel priming pump is DC powered, but incorrect because the fuel priming pump is only needed during DG start and stops after engine speed exceeds 870 RPM.

Distractor 3 is plausible because the DC soak back pump is powered by DC, but incorrect because the DC Soak back pump is the backup to the normally operating AC soak back pump.

Reference: LOA-DC-201, Rev 24; LP 011 EDG and Auxiliaries, Rev 19
Reference provided during examination: None

Cognitive level: High
10CFR Part 55: 41.7

Importance Level (RO/SRO): 4.0
Tier: 2 Group: 1

PRA: Yes
K/A: 264000 K6.09 - Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF6 EGE)
EMERGENCY GENERATORS (DIESEL/JET): DC electrical distribution system

Safety Function: 6
SRO Justification: N/A

Question Source: New

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Question History:

Comments:

Objective #: 011.00.13

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

50

ID: 2544694

Points: 1.00

Unit 1 Is at 100% RTP.

Instrument air header pressure is 115 psig and lowering at 1 psig/min.

Assuming NO operator action, what is the HIGHEST pressure that the standby air compressor will automatically start?

- A. 80 psig
- B. 90 psig
- C. 100 psig
- D. 110 psig

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: " 100 psig " is the correct answer.

Per LOA-IA-101

ECs #344407, #344408, and #344409 have installed a new microprocessor control panel the three SACs. With these new controls if a compressor is in Standby Mode, it should auto start and load with the receipt of a low station air header pressure (@ ≈100 psig). It also will automatically respond to Surge Events by unloading and then reloading the air compressor. If this occurs 4 times in 10 minutes, the new controller will unload the compressor until operator action is taken to restore it or until 20 minutes elapses with the SAC running unloaded at which point the SAC will shutdown. EC 628041 added a fourth SAC, 0SA04C. These EC also installed new Control Room control switches (EC #352455 did this for Unit 1). These allow the operator to manually start or stop an air compressor from the Control Room.

Distractor 1 is plausible because the reactor building low instrument air pressure alarm occurs at 80 psig, but incorrect because the standby air compressor will have started and loaded at 100 psig.

Distractor 2 is plausible because 110 psig is the bottom of the normal operating band for the SA compressor, but incorrect because the SA compressor control system is designed to auto start at a SA header pressure of 100 psig.

Distractor 3 is plausible because 90 psig is the low alarm pressure for IA receiver, but incorrect because the SA compressor control system is designed to auto start at a SA header pressure of 100 psig,

Reference: LOA-IA-101, Rev 18; LOR-1PM10J-B204, Rev 1

Reference provided during examination: None

Cognitive level: Memory

10CFR Part 55: 41.7

Importance Level (RO/SRO): 3.5

Tier: 2 Group: 1

PRA: Yes

K/A: 300000 A3.03 - Ability to monitor automatic operation of the (SF8 IA) INSTRUMENT AIR SYSTEM including: Compressor automatic starts/trips

Safety Function: 8

SRO Justification: N/A

Question Source: New

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Question History:

Comments:

Objective #: 120.00.10

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

51

ID: 2544727

Points: 1.00

Unit 1 is at 100% RTP.

- Flow indicator meter orifices for the RWCU pump heat exchangers were replaced
- Original orifice openings were found eroded (slightly larger)
- ACTUAL system flow was not changed after replacement

On return to service, INDICATED RWCU pump heat exchanger flow is ____ (1) ____ than before replacement due to a ____ (2) ____ D/P across the orifice. (Assume all other parameters remain constant)

- A. (1) lower
(2) larger
- B. (1) lower
(2) smaller
- C. (1) higher
(2) larger
- D. (1) higher
(2) smaller

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "(1) higher, (2) larger" is the correct answer.

When a new orifice plate is installed without changing the set flow rate the indicated flow will be higher than the indicated flow of the previous worn orifice due to the larger D/P across the smaller opening.

Per GEFES BWR / COMPONENTS / CHAPTER 7 / SENSORS & DETECTORS

The basic principle is that when a fluid flows through a restriction, its velocity will increase and its pressure will decrease. Although the fluid's velocity increases, its volumetric flow rate (V) does not change. The change in pressure (D/P), however, is proportional to the square of the volumetric flow rate. $D/P \propto (V)^2$

Therefore, since D/P is measured, the volumetric flow rate can be determined.

Orifice plates are the most common type of primary flow element because they are inexpensive and easy to install. However, orifice plates have two serious disadvantages: (1) a high permanent pressure drop, and (2) erosion of the sharp edges.

Distractors 1, 2 and 3 are plausible because they are possible results depending on configuration and homogeneous, but incorrect because the change to the new orifice with a smaller sized hole will indicate a higher flowrate due to a larger D/P.

Reference: GEFES BWR / COMPONENTS / CHAPTER 7 / SENSORS & DETECTORS, Rev 4c, Drawing M-90, Sheet 3

Reference provided during examination: None

Cognitive level: High
10CFR Part 55: 41.7

Importance Level (RO/SRO): 2.5
Tier: 2 Group: 1

PRA: No

K/A: 400000 (SF8 CCS) COMPONENT COOLING WATER SYSTEM (291002 K1.01) SENSORS AND DETECTORS (FLOW) Operational characteristics of venturis and orifices

Safety Function: 8
SRO Justification: N/A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Question Source: New

Question History:

Comments:

Objective #: 291002

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

52

ID: 2544745

Points: 1.00

Unit 2 is in STARTUP at 7% RTP with the Motor Driven Reactor Feed Pump (MDRFP) running.

Service water flow to Turbine Building Closed Cooling Water (WT) is lost.

Assuming NO operator action, what will be the impact on the operation of the MDRFP?

- A. Failure of the thrust bearing
- B. Alarm for high lube oil PRESSURE
- C. Trip on high lube oil TEMPERATURE
- D. Trip on high thrust bearing vibration

Answer: A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Failure of the thrust bearing" is the correct answer.

A Loss of Service Water to Turbine Building Closed Cooling Water which provides cooling for the MDRFP lube oil and seals will cause overheating and possible failure of the bearings and/or seals.

Per UFSAR 9.2.8.2

The TBCCW system provides cooling water to the motor-driven reactor feed pump mechanical seal heat exchangers, mechanical seal jacket cooler cavities and oil coolers.

Distractor 1 is plausible because lube oil will be affected and there is an oil pressure alarm, but incorrect because there is no high lube oil pressure alarm, only an alarm and trip for low pressure.

Distractor 2 is plausible because some equipment trips on high oil temp such as station air compressors, but incorrect because there is no Hi lube oil temperature trip associated with MDRFP.

Distractor 3 is plausible because the high temperatures will likely cause bearing damage and the TDRFPs have a high thrust bearing vibration trip, but incorrect because there is no high thrust bearing vibration trip associated with MDRFP.

Reference: UFSAR 9.2.8.2, LOR-1PM03J-A301, Rev 1; LP 077 Feedwater System, Rev 22

Reference provided during examination: None

Cognitive level: Memory

10CFR Part 55: 41.5

Importance Level (RO/SRO): 2.7

Tier: 2 Group: 1

PRA: Yes

K/A: 510000 K1.13 - Knowledge of the physical connections and/or cause and effect relationships between the (SF4 SWS*) SERVICE WATER SYSTEM and the following systems: Feedwater system

Safety Function: 4

SRO Justification: N/A

Question Source: New

Question History:

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Comments:

Objective #: Objective 077.00.13

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

53

ID: 2505282

Points: 1.00

Unit 2 is at 100% RTP.

A "NON-critical self-test" fault within the Rod Control Management System (RCMS) Controller occurs.

What is the response, if any, of the RCMS system?

- A. NO RCMS alarms annunciate
- B. RCMS INOPERATIVE Alarm ONLY.
- C. RCMS TROUBLE / TEST Alarm ONLY.
- D. RCMS INOPERATIVE and RCMS TROUBLE / TEST Alarms

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer explanation: "RCMS TROUBLE / TEST Alarm ONLY." is the correct answer.

Per LOR-1H13-P603-A402, RCMS TROUBLE / TEST Discussion section step 3.f.

The RCMS Trouble/Test Alarm is activated upon a Non-Critical Cross-Compare failure with the other RCMS Controller

Distractor 1 is plausible because non-critical self-test faults are minor issues, but incorrect because a Non-critical self-test fault generates a RCMS trouble alarm.

Distractor 2 is plausible because this a potential alarm for self-test fault, but incorrect because this would be true if it was a critical self-test fault.

Distractor 3 is plausible because these alarms generated in the RCMS system for a self-test fault, but incorrect because the inoperative alarm would only occur if it was a critical self-test fault.

Reference: LOR-1H13-P603-A502 RCMS INOPERATIVE rev 7; LOP-RM-03 ROD CONTROL MANAGEMENT SYSTEM EVENT AND ALARM MESSAGES rev 8; LOR-1H13-P603-A402, RCMS TROUBLE / TEST, rev 11
LP#047 Rod Control Management System (RCMS)

Reference provided during examination: none

Cognitive level: Low

10CFR Part 55: 41.5

Importance Level (RO/SRO): 3.3

Tier: 2 Group: 2

PRA: No

K/A: 201002 A1.06 Ability to predict and/or monitor changes in parameters associated with operation of the Reactor Manual Control System, including: Lights and alarms

Safety Function: 1 – Reactivity Control

SRO Justification: N/A

Question Source: Bank

Question History:

Comments:

Objective #: 047.00.16.004

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

54

ID: 2505286

Points: 1.00

Unit 1 was at 100% RTP when an ATWS occurred.

During performance of LGA-NB-101, Alternate Rod Insertion, Single Rod Insertion, the operator is directed to place the RWM Mode Select switch in Bypass for the Rod Worth Minimizer.

This action bypasses ...

- A. rod insert blocks to allow inward rod motion.
- B. the settle function to speed the rate of rod insertion.
- C. the single notch function to speed the rate of rod insertion.
- D. nuclear instrumentation rod blocks to allow all rod motion.

Answer: A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "rod insert blocks to allow inward rod motion." is the correct answer.

The RWM Bypass/Run switch has two redundant contacts fed to each channel of the MCR Interface. Positioning this switch to BYPASS will remove all RWM insert and withdraw block contributions from the logic of both RCMS Controller Channels.

Per LGA-NB-01 Step E.5.d and Attachment J, placing the RWM Mode Select switch in Bypass will bypass insert rod blocks.

Distractor 1 is plausible because this will speed rod insertion, but incorrect because this function is completed by the continuous insert push button and not the RWM bypass.

Distractor 2 is plausible because this will speed rod insertion, but incorrect because this function is completed by the continuous insert push button and not the RWM bypass.

Distractor 3 is plausible because bypass switch does bypass rod blocks, but incorrect because it only bypasses rod blocks generated by the RWM.

Reference: LGA-NB-01 E.5.d, rev 0, LP 047 Reactor Control Management System (RCMS) , Rev 5

Reference provided during examination: None

Cognitive level: Memory

10CFR Part 55: 41.5

Importance Level (RO/SRO): 3.5

Tier: 2 Group: 2

PRA: No

K/A: 201006K5.13 - Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF7 RWMS) ROD WORTH MINIMIZER SYSTEM: Insert Block

Safety Function: 7

SRO Justification: N/A

Question Source: Bank

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Question History:

Comments:

Objective #: 047.00.05

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

55

ID: 2553949

Points: 1.00

Unit 2 is at 100% RTP

The Mixer on Jet pump #5 becomes displaced.

Actual core flow will...

- A. lower due to higher back pressure on 2A RR pump.
- B. lower due to reverse flow through the displaced mixer.
- C. rise due to lower back pressure on 2B RR pump.
- D. rise due to lower flow restriction from the displaced mixer.

Answer: B

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer explanation: " lower due to back flow through the displaced mixer" is the correct answer.

Per LOA-RR-201 discussion section C.12, a displaced mixer will cause the associated jet pump to lose drive flow and the displaced jet pump will start reverse flow. Core power and electrical output will lower due to a loss in core flow. Core flow indication will rise due to the addition of the reverse flow to the core flow indication.

Distractor 1 is plausible because core flow will lower, but incorrect because core flow lowers due to reverse flow through the displaced mixer.

Distractor 2 is plausible because flow through the 2B RR pump will rise, but incorrect because the drop in flow through the 2A RR loop jet pumps is greater than the flow added by the unaffected loop jet pumps.

Distractor 3 is plausible because the mixer section of the jet pump raises pressure and a drop in backpressure in a system with a centrifugal pump will cause flow to go up, but incorrect because the lower pressure will be overcome by the higher pressure in the core inlet plenum and backflow will occur through the displaced mixer.

Reference: LOA-RR-201 rev 48; UFSAR Appendix G

Reference provided during examination: None

Cognitive level: Memory

10CFR Part 55: 41.5

Importance Level (RO/SRO): 4.2

Tier: 2 Group: 2

PRA: No

K/A: 202001 K3.01 - Knowledge of the effect that a loss or malfunction of the (SF1, SF4 RS) RECIRCULATION SYSTEM will have on the following systems or system parameters: Core flow

Safety Function: 1

SRO Justification: N/A

Question Source: New

Question History:

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Comments:

Objective #: 022.00.16

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

56

ID: 2505287

Points: 1.00

Unit 1 is at 60% RTP when the 1A FCV drifts full open.

(1) What is the most likely condition to occur?

(2) What operator action is required to prevent that condition?

- A. (1) Core Instability Event
(2) INSERT CRAM arrays ONLY
- B. (1) Core Instability Event
(2) REDUCE power and downshift both RR pumps
- C. (1) Jet Pump Vibration
(2) INSERT CRAM arrays ONLY
- D. (1) Jet Pump Vibration
(2) REDUCE power and downshift both RR pumps

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "(1) Jet Pump Vibration (2) REDUCE power and downshift both RR pumps" is the correct answer.

At 60% RTP total RR flow is approximately 60 Mlbm/hr split between the two RR loops, about 30 Mlbm/hr each, with a rated flow of about 108.5Mlbm/hr. When the 1A FCV fails full open the 1A RR loop flow will rise close to 54.25 Mlbm/hr. The difference in the loop flows will be greater than the least restrictive 10.85 Mlbm/hr which will place the loop flow mismatch greater than the analyzed mismatch for jet pump components. The jet pumps were not designed for the large horizontal flow loading. Unbalanced loop flow will induce jet pump vibration.

Per LOA-RR-101, Discussion C.4.3, Recirculation loop flows should be rebalanced as soon as practical to minimize jet pump vibration.

LOA-RR-101 section B.10 provides options to restore loop flow to less than TS Mismatch by, opening the low flow FCV if operable, OR REDUCING power per LGP-3-1 including shifting both RR pumps to slow speed, OR Tripping the low flow RR pump.

Distractor 1 is plausible because core flux will be affected and CRAM arrays are designed to exit region 1, but incorrect because core instabilities result from flow reduction into the instability region, not flow increases.

Distractor 2 is plausible because reducing power including shifting both RR pumps to slow speed will restore the loop mismatch to less than the TS required which is associated with the loop flow mismatch concern as both RR FCV will be fully opened following the downshift, but incorrect because the concern with loop flow mismatch is failure of Jet pump components due to vibrations caused by the mismatch.

Distractor 3 is plausible because Jet pump vibrations are the concern with mismatched loop flows, but incorrect because lowering power using rods will not restore the loop flow mismatch.

Reference: LOA-RR-101, Rev 46; LOP-RR-05, Rev 56

Reference provided during examination: None

Cognitive level: High

10CFR Part 55: 41.5

Importance Level (RO/SRO): 3.8

Tier: 2 Group: 2

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

PRA: No

K/A: 202002 A2.04 - Ability to (a) predict the impacts of the following on the (SF1 RSCTL) RECIRCULATION FLOW CONTROL SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Recirculation pump speed/loop flow mismatch

Safety Function: 1

SRO Justification: N/A

Question Source: New

Question History:

Comments:

Objective #: 022.00.16

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

57

ID: 2505292

Points: 1.00

Unit 1 is operating at 100% RTP.

1C Reactor Water Cleanup (RT) Filter Demin flow lowers to 30 gpm and stabilizes.

How does the RT system respond?

- A. The F/D remains IN-SERVICE, Hold Pump Outlet valve OPENS, Hold Pump STARTS.
- B. Hold Pump Starts, Hold Pump Outlet valve OPENS, The F/D ISOLATES, F/D Outlet valve CLOSES.
- C. The F/D Outlet valve CLOSES, Hold Pump Outlet valve Closes, Hold Pump Trips, Filter Demin ISOLATES.
- D. The F/D Inlet valve CLOSES, Hold Pump Outlet valve OPENS, Hold Pump Starts, F/D Outlet valve OPENS.

Answer: B

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Hold Pump Starts, Hold Pump Outlet valve OPENS; The F/D ISOLATES, Outlet valve CLOSES." is the correct answer.

Per LOP-RT-04, D.2.3

The Filter/Demineralizer Low Flow Trip:

- A Filter/Demineralizer at 34 gpm.
- B Filter/Demineralizer at 47 gpm
- C Filter/Demineralizer at 61 gpm.

The Filter/Demineralizer receiving Low Flow trip will automatically transfer to HOLD Mode where the affected Filter Demineralizer outlet valve, 1(2)G33-Z001-66A/B/C will close.

Per LOP-RT-04, Step E.5.1.3

As Filter Demineralizer flow lowers the

- Hold Pump 1(2)G33-Z001-3A/B/C starts,
- the Holding Valve 1(2)G33-Z001-14A/B/C HA/HB/HC opens,
- Vessel Outlet Valve 1(2)G33-Z001-66A/B/C closes, and the
- Hold Light comes on

Distractor 1 is plausible because this is possible system configuration, but incorrect because the F/D isolates at 61 gpm. This condition is only exists with F/D flowrates <100 gpm and > 61 gpm.

Distractor 2 is plausible because this is a possible system configuration, but incorrect because the Hold pump starts at 100 gpm decreasing and continues to run.

Distractor 3 is plausible because this is a possible system configuration, but incorrect because the F/D will isolate and close the F/D outlet valve.

Reference: LOP-RT-04, Rev 22; LOA-RT-101, Rev 23; LP 027 Reactor Water Clean-up, Rev 12

Reference provided during examination: None

Cognitive level: Memory

10CFR Part 55: 41.7

Importance Level (RO/SRO): 3.2

Tier: 2 Group: 2

PRA: No

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

K/A: 204000 A3.07 - Ability to monitor automatic operation of the (SF2 RWCU)
REACTOR WATER CLEANUP SYSTEM including: System flow

Safety Function: 2

SRO Justification: N/A

Question Source: Bank

Question History:

Comments:

Objective #: 027.00.10

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

58

ID: 2505293

Points: 1.00

Unit 1 is at 100% RTP.

A 5 gpm leak develops on the common variable leg for Narrow Range Level transmitters 1B and 1C.

If Reactor water level rises to the level 8 trip setpoint, the Main Turbine will ...

- A. trip because Main Turbine Level 8 trip logic is backed up by RCIC instruments.
- B. trip because Main Turbine Level 8 logic requires 1-out-of-3 instruments to actuate.
- C. trip because Main Turbine Level 8 trip logic is from the Wide Range Level instruments.
- D. NOT trip because Main Turbine Level 8 trip logic requires 2-out-of-3 instruments to actuate.

Answer: A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

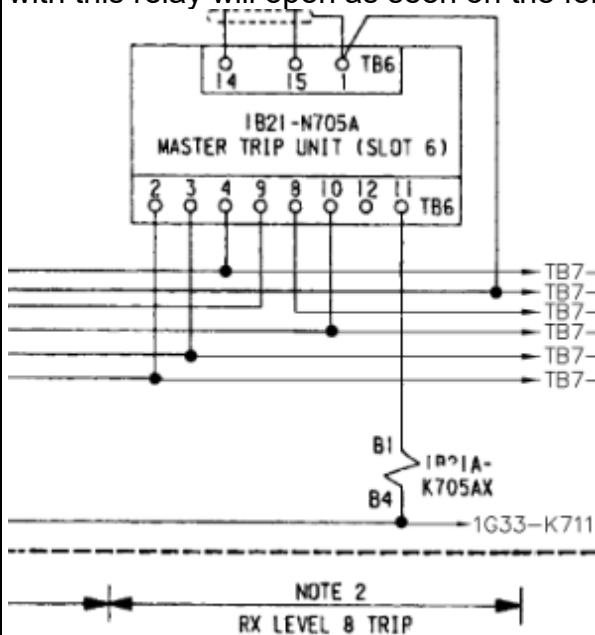
LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "trip because Main Turbine Level 8 trip logic is backed up by RCIC instruments." is the correct answer.

The 1A RCIC level transmitter has been tapped as a backup for the level 8 turbine and feed pump trip. This design was added to ensure that there was a backup level 8 trip in the unlikely event that a variable leg leak occurs on these instruments.

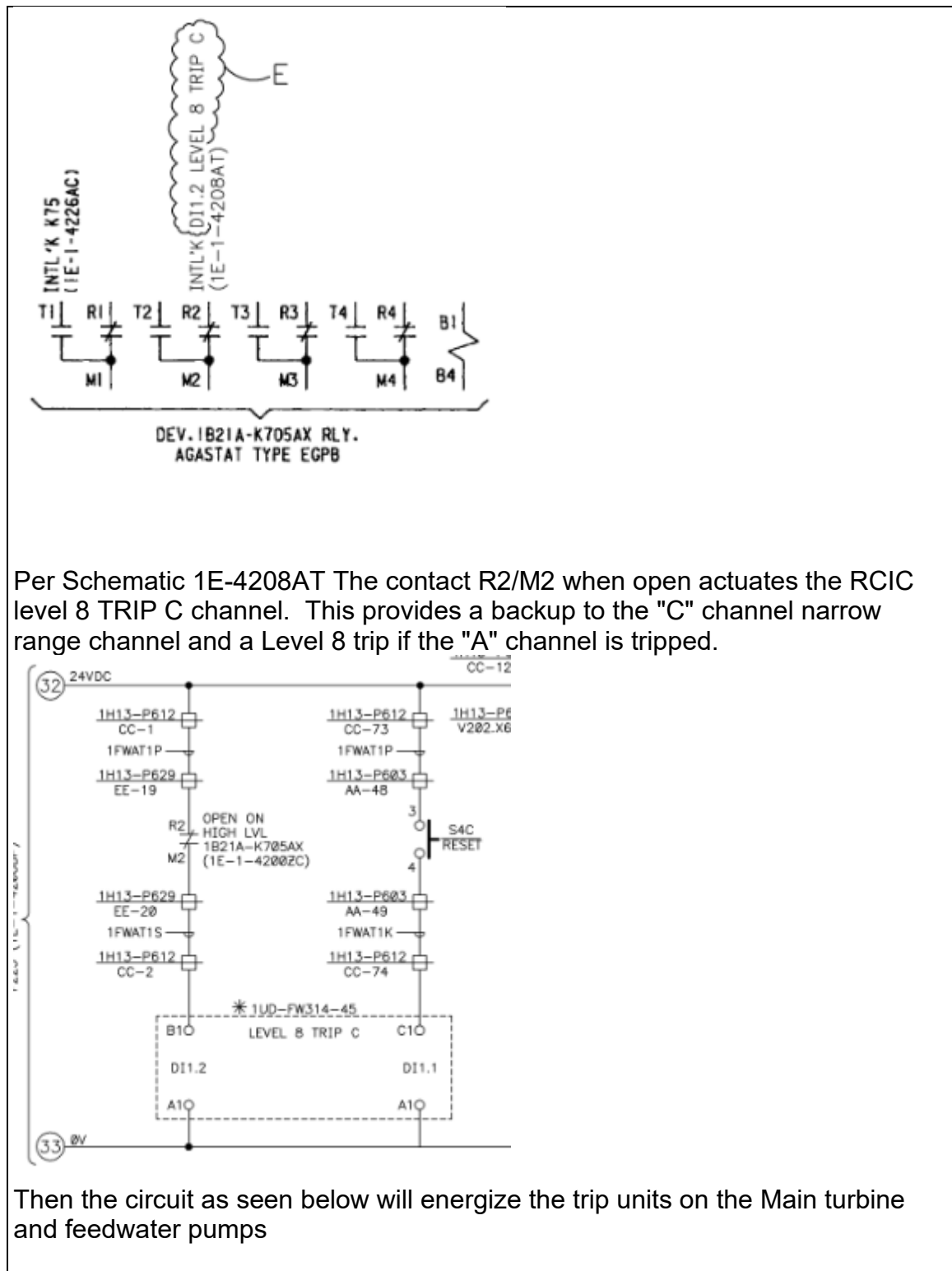
Per Schematic 1E-1-4200ZC this depicts master trip Unit 1B21-N705A and if a High level trip relay 1B21-K705AX will energize. The contacts R2/M2 associated with this relay will open as seen on the following schematic 1E-1-4208AT



Test Answer Key

LaSalle ILT 23-1 NRC RO

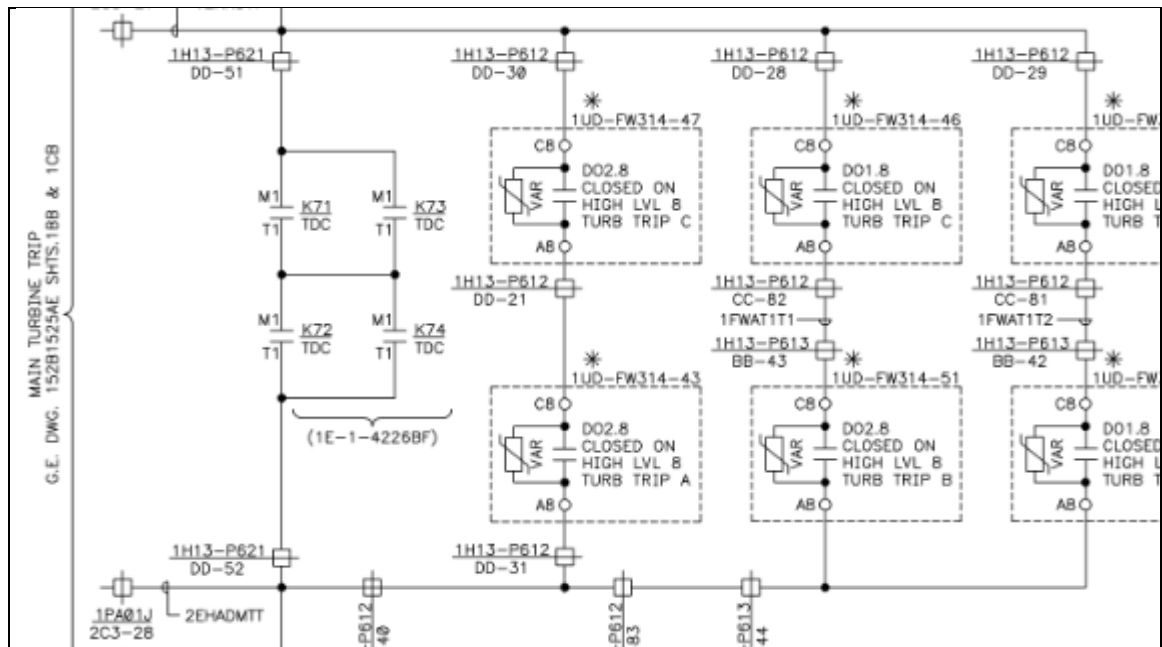
Test ID: 375864



Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864



Distractor 1 is plausible because it is set up 2 out of 3, but incorrect because adding the RCIC channel to one of the trip channels has added more redundancy in the system and under this scenario the system will still trip.

Distractor 2 is plausible because reactor water level interlocks are tripped from the wide range instruments, but incorrect because the narrow ranges provide the source of trip input.

Distractor 3 is plausible because the turbine will still trip, but incorrect because the logic is 2 out of 3 supported by the backup trip of the 'C' channel from RCIC.

Reference: Schematic 1E-1-4200ZC; 1E-1-4208AT; LOR-1H13-P603-A310, Rev 7; Feedwater Control Reactor Vessel Water Trip - Level 8; System Lesson Plan #040 Reactor Vessel Instrumentation, Rev 10
Reference provided during examination: None

Cognitive level: High
10CFR Part 55:41(b)(7)

Level (RO/SRO): 3.3
Tier: 2 Group: 2

PRA: No

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

K/A: 216000 K6.05 - Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF7 NBI) NUCLEAR BOILER INSTRUMENTATION: Instrument channels

Safety Function: 7

SRO Justification: N/A

Question Source: Bank

Question History:

Comments:

Objective #040.00.16

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

59

ID: 2545142

Points: 1.00

Unit 2 was at 100% RTP when a LOCA occurred.

- Drywell pressure is 32 psig and rising 0.5 psig/min
- Suppression Chamber pressure is 27 psig and rising 0.5 psig/min
- Suppression Pool water level is 701 ft and rising 0.1 ft/min

IAW OP-AA-103-102-1001, STRATEGIES FOR SUCCESSFUL TRANSIENT MITIGATION, what is the required NSO report to the Unit Supervisor?

- A. Drywell pressure is 32 psig rising 0.5 psig/min.
- B. Drywell pressure is below PCPL but rising rapidly.
- C. Primary Containment pressure is 32 psig and rising.
- D. Drywell pressure is greater than the peak PSP curve pressure and rising.

Answer: A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Drywell pressure is 32 psig rising 0.5 psig/min." is the correct answer.

The operator should concisely report status including critical information with enough detail to validate any conclusion regarding the status of primary containment. Communication of parameters only reduces the possibility that all hearing the report will understand that Primary containment is not functioning correctly.

Per OP-LA-103-102-1002,

Updates should be used when one individual on the crew has a piece of information that will benefit all other crewmembers. Examples include but are not limited to:

- Degrading plant parameter
- Approaching critical plant parameters i.e. any EOP entry conditions, Top of active fuel, etc.

Per OP-AA-103-102-1001, The communication of parameters shall include the indicated parameter value, units of the measured parameter and trend including rate.

Distractor 1 is plausible because the Drywell pressure is below PCPL, which will require additional action if violated, and rising rapidly, but incorrect because it does not provide the parameter value and units and Drywell pressure is normally below the PCPL.

Distractor 2 is plausible because the drywell is part of primary containment and is 32 psig and rising 0.5 psig/min, but incorrect because the Drywell and Suppression Chamber are both part of primary containment.

Distractor 3 is plausible because containment pressure is above the peak PSP pressure of 28.6 psig, but incorrect because suppression chamber pressure is used to determine PSP violation.

Reference: OP-LA-103-102-1002, Rev 26 ; LGA-003 Rev 19

Reference provided during examination: None

Cognitive level: Memory

10CFR Part 55: 41.10

Importance Level (RO/SRO): 3.6 / 3.8

Tier: 2 Group: 2

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

PRA: No

K/A: 223001 G2.1.18 - (SF5 PCS) PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES / CONDUCT OF OPERATIONS Ability to make accurate, clear, and concise logs, records, status boards, and reports

Safety Function: 5

SRO Justification: N/A

Question Source: New

Question History:

Comments:

Objective #: 090.00.11

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

60

ID: 2545163

Points: 1.00

Unit 1 is at 100% RTP when power to the Main Steam Aux Supply Stop Valve, 1B21-F418A, is lost.

The 1B21-F418A valve will ____ (1) ____ and the 1A TDRFP will ____ (2) ____ .

- A. (1) close
(2) trip
- B. (1) close
(2) continue running
- C. (1) remain open
(2) trip
- D. (1) remain open
(2) continue running

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "(1) remain open; (2) continue running" is the correct answer.

Per 1E-1-4203AN, 1B21-F418A is a motor operated valve that will fail as-is on a loss of MCC 136X-2, compt D4. The 1A TDRFP receives high pressure steam through the 1B21-F418A valve but will be unaffected because the valve will stay open and also because at high power levels the steam supplied to TDRFP's comes from the main turbine cross-over piping through the Low pressure control valves.

Distractor 1 is plausible because some valves fail close on a loss of power and the TDRPF would trip will trip with a loss of HP steam at low power levels, but incorrect because 1B21-F418A is an MOV which will fail as-is and the TDRFP will continue running.

Distractor 2 is plausible because it would be true if the valve failed closed, but incorrect because 1B21-F418A is an MOV which will fail as-is.

Distractor 3 is plausible because MCC 136X-2 powers MSIV solenoid which will lose power and potentially close the MSIVs, but incorrect because even a loss of the entire bus would only place the MSIVs in a half trip condition so steam flow would not be lost.

Reference: Drawing 1E-1-4203AN; 1E-1-4203AB, 1E-1-4203AF; 1E-1-4000CW; 1E-1-4000LJ; Lesson Plan 077 Feedwater Rev. 22

Reference provided during examination: None

Cognitive level: High
10CFR Part 55: 41.7

Importance Level (RO/SRO): 3.3
Tier: 2 Group: 2

PRA: Yes
K/A: 239001 K2.02 - (SF3, SF4 MRSS) MAIN AND REHEAT STEAM SYSTEM
Knowledge of electrical power supplies to the following: Main steam line shutoff valves

Safety Function: 3
SRO Justification: N/A

Question Source: New
Question History:

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Comments:

Objective #: 070.00.13

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

61

ID: 2505299

Points: 1.00

Unit 1 is at 100% power when the Main Turbine thrust bearing fails.

What is an expected action because of this failure?

- A. Generator Field breaker opens.
- B. Turbine Control Valve drains close.
- C. Extraction steam dump valves close.
- D. Feedwater Heaters Non-Return check valves open.

Answer: A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Generator Field breaker opens." is the correct answer.

Per LOA-TG-101 and LP 071 Main Turbine and Auxiliary System, the following are verified to occur on a turbine trip

- All MSVs and TCVs close
- TCV drains open
- OCBs 9-10 and 10-11 open
- Generator field breaker opens
- Feedwater heater Non-return check valves close
- FW heater spill (dump) valves open

Distractor 1 is plausible because turbine control valve drains are affected on a turbine trip, but incorrect because the valves open when the turbine trips.

Distractor 2 is plausible because Extraction steam dump valves are affected on a turbine trip, but incorrect because the valves open when the turbine trips.

Distractor 3 is plausible because FW heater non-return check valves are affected on a turbine trip, but incorrect because the valves close when the turbine trips.

Reference: LOA-TG-101, Rev 25; LP 071 Main Turbine and Auxiliary System, Rev 14

Reference provided during examination: None

Cognitive level: High
10CFR Part 55: 41.7

Importance Level (RO/SRO): 3.5
Tier: 2 Group: 2

PRA: Yes

K/A: 245000 K4.06 - Knowledge of (SF4 MTGEN) MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS design features and/or interlocks that provide for the following: Generator protection

Safety Function: 4
SRO Justification: N/A

Question Source: Bank
Question History:

Comments: Objective #: 071.00.14

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

62

ID: 2545211

Points: 1.00

Unit 2 is at 100% RTP when a Primary Containment Ventilation malfunction results in lowering drywell temperature by 30°F.

As drywell temperature lowers, INDICATED narrow range level will be (1) than ACTUAL level because the reference leg would be (2) dense.

- A. (1) lower
(2) less
- B. (1) lower
(2) more
- C. (1) higher
(2) less
- D. (1) higher
(2) more

Answer: B

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: “(1) lower, (2) more “ is the correct answer.

As drywell temperature lowers, the reference leg fluid will become more dense. As the reference leg becomes more dense the differential pressure sensed by the level transmitter will become greater, causing indicated level to lower.

Per GEFS BWR / COMPONENTS / CHAPTER 7 / SENSORS & DETECTORS, Rev 4c

Temperature variations can significantly affect level measurement accuracies. An increase in ambient temperature causes the density of the reference leg to decrease. This will result in a lower D/P sensed by the D/P cell, and indicated level will be greater than actual level. The opposite effect produces a lower indicated level when the ambient temperature decreases.

Distractor 1 is plausible because DW temperature change will impact reference leg density, but incorrect because liquids become more dense at lower temperatures.

Distractor 2 is plausible because indicated level will be affected, but incorrect because a more dense wet reference leg will have a higher Dp and a lower indicated level.

Distractor 3 is plausible because indicated level will be affected and DW temperature change will impact reference leg density, but incorrect because liquids become more dense at lower temperatures and a more dense wet reference leg will have a higher Dp and a lower indicated level.

Reference: P and ID M-93 Sheet 5; LOA-NB-101 REACTOR VESSEL INSTRUMENTATION ABNORMAL Rev. 12; GEFS BWR / COMPONENTS / CHAPTER 7 / SENSORS & DETECTORS, Rev 4c

Reference provided during examination: none

Cognitive level: High
10CFR Part 55:41.7

Level (RO/SRO): 2.9
Tier: 2 Group: 2

PRA: No

K/A: 291002 K1.08 / 288000 - (SF9 PVS) PLANT VENTILATION SYSTEMS / SENSORS AND DETECTORS (LEVEL) Effects of operating environment (pressure, temperature, or radiation)

Safety Function: 9

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

SRO Justification: N/A

Question Source: Bank

Question History:

Comments:

Objective # 040.00.11

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

63

ID: 2546431

Points: 1.00

What is a function of the Reactor Vessel Core Support Plate?

The Reactor Vessel Core Support Plate provides ...

- A. LATERAL support to the Jet Pumps.
- B. VERTICAL support to the Steam Dryers.
- C. LATERAL support to the CRD Guide Tubes.
- D. VERTICAL support to the IRM guide tubes.

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Answer Explanation: "LATERAL support to the CRD Guide Tubes." is correct.

Per UFSAR 3.9.5 The Reactor Vessel Core Support Plate provides lateral support for the CRD Guide Tubes.

Distractor 1 is plausible because jet pumps are supported laterally by a plate between the downcomer annulus and the core inlet plenum, but incorrect because this is the baffle support plate.

Distractor 2 is plausible because the steam dryers are located above the reactor core and other components, such as fuel rods, rest on the core plate and transmit their weight to the reactor vessel bottom head, but incorrect because the steam dryers are supported by brats attached to the Reactor vessel wall.

Distractor 3 is plausible because the Reactor Core plate provides lateral support to the IRM guide tubes, but incorrect because only the peripheral fuel supports and the neutron sources are supported vertically by the reactor core plate.

Reference: UFSAR 3.9.5

Reference provided during examination: None

Cognitive Level: Memory

10CFR Part 55: 41(b)(2)

Importance Level (RO/SRO): 3.7

Tier: 2 Group: 2

PRA: No

Safety Function: 4 Heat Removal from the Reactor Core

K/A: 290002 REACTOR VESSEL INTERNALS K1.11 Knowledge of the physical connections and/or cause and effect relationships between the Reactor Vessel and Internals and the following systems: Control rod drive mechanism system

SRO Justification: N/A

Question Source: Bank

Question History:

Comments:

Objective #: 020.00.04

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

64

ID: 2504910

Points: 1.00

A locked manual valve in a safety-related system is to be verified locked in the OPEN position.

Assume locking device will allow some valve movement.

Which of the following is required IAW OP-AA-108-101-1001, COMPONENT POSITION DETERMINATION?

INITIALLY, attempt to move the valve in the ____ (1) ____ direction with the locking device ____ (2) ____.

- A. (1) CLOSED
(2) INSTALLED
- B. (1) CLOSED
(2) REMOVED
- C. (1) OPEN
(2) INSTALLED
- D. (1) OPEN
(2) REMOVED

Answer: A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Answer Explanation: “ (1) CLOSED, (2) INSTALLED “ is the correct answer.

Per OP-AA-108-101-1001 "COMPONENT POSITION DETERMINATION" states as follows step 4.3 "ENSURE that the locking device is mounted securely to prevent valve movement and that the padlock is in a locked position. For a locked open valve, ATTEMPT to move the valve in the closed direction to ensure the locking device maintains the valve open."

Distractor 1 is plausible because some situations require removal of the locking device, but incorrect because some movement is possible per the stem.

Distractor 2 is plausible because checking in the open direction could be used, but incorrect because for open valves the valve must be checked in the closed direction.

Distractor 3 is plausible because some situations require removal of the locking device and checking in the open direction could be used, but incorrect because some movement is possible and for open valves the valve must be checked in the closed direction.

Reference: OP-AA-108-101-1001 COMPONENT POSITION DETERMINATION rev 4

Reference provided during examination: None

Cognitive level: Memory
10CFR Part 55: 41.10

Importance Level (RO/SRO): 4.1
Tier: 3 Group: N/A

PRA: No
K/A: G2.1.29 - CONDUCT OF OPERATIONS Knowledge of how to conduct system lineups, such as valves, breakers, or switches

Safety Function: N/A
SRO Justification: N/A

Question Source: Bank
Question History: ILT 13-1 NRC

Comments:
Objective #: LAS-LO-639

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

65

ID: 2504911

Points: 1.00

Who is responsible for ensuring that core unloading is performed per the approved Nuclear Component Transfer List with each step of completion?

- A. Shift Manager
- B. Reactor Operator
- C. Qualified Nuclear Engineer
- D. Nuclear Materials Custodian

Answer: B

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Reactor Operator" is the correct answer.

Per LFP-100-1 MASTER REFUEL PROCEDURE.

- E.3.7 VERIFY operability of Direct (continuous) Communications between Control Room and Refueling Platform once per 12 hours. TRM TSR 3.9.b.1.
- E.3.8 An NSO assigned to CORE ALTERATIONS shall CHECK the Step for correctness. IF it is in agreement with NCTL, THEN the NSO gives the Fuel Handling personnel permission to proceed, per Attachment A.

Distractor 1 is plausible because the Shift Manager provides oversight for Operations supervisors and provides approval to defeat protective features in LFP-100-1, but incorrect because an NSO assigned to CORE ALTERATIONS gives permission to fuel handling personnel to proceed at every step.

Distractor 2 is plausible because QNEs Nuclear Engineering develops the Nuclear Component Transfer List, but is incorrect because an NSO assigned to CORE ALTERATIONS gives permission to fuel handling personnel to proceed at every step.

Distractor 3 is plausible because the Nuclear Fuels Custodian is responsible for retaining the NCTL, but incorrect because the Nuclear Materials Custodian maintains the records for Nuclear Material and is not required to ensure the material is loaded correctly IAW the NCTL.

Reference: LFP-100-1 MASTER REFUEL PROCEDURE rev 70; LFP-100-2 rev 16; OP-AA-101-111 rev 15

Reference provided during examination: None

Cognitive level: Memory
10CFR Part 55:41.10

Importance Level (RO/SRO): 3.9 / NA
Tier: 3 Group:

PRA: No

K/A: G 2.1.44 Knowledge of RO duties in the control room during fuel handling, such as

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

responding to alarms from the fuel handling area, communication with fuel handling personnel, operating systems from the control room in support of fueling operations, or supporting instrumentation

Safety Function: 1

SRO Justification: N/A

Question Source: Bank

Question History:

Comments:

Objective #:301.00.05

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

66

ID: 2550538

Points: 1.00

Unit 1 and Unit 2 are at 100% RTP.

Which Technical Specification limit is different between Unit 1 and Unit 2?

- A. Reactor Core Safety Limit
- B. Minimum Critical Power Ratio
- C. Reactor Coolant System Pressure Safety Limit
- D. Rod Block Monitor Upscale Trip Allowable Value

Answer: B

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer explanation: "MCPR" is the correct answer.

Per COLR for LaSalle Unit 1 Cycle 21 the MCPR limit for 100% RTP is 1.37 (Option A) and 1.31 (Option B). Per COLR for LaSalle Unit 2 Cycle 20 the MCPR limit for 100% RTP is 1.33 (Option A) and 1.30 (Option B). Options are based on scram speeds meeting testing requirements and any combination can exist between the two Units, but no combination will result in the MCPR limit being the same between Unit 1 and Unit 2.

Distractor 1 is plausible because MCPR is one of the Reactor Core Safety Limits, but incorrect because the MCPR value listed in the safety limit is the most limiting of MCPR values for fuel product lines loaded in the core and is the same for Unit 1 and Unit 2.

Distractor 2 is plausible because the Reactor Coolant Pressure boundary is subjected to radiation exposure and is monitored using sample coupons installed in the core that are periodically removed and tested which could show a degradation in the integrity of the pressure boundary, but incorrect because the pressure boundary safety limit is the same between Unit 1 and Unit 2.

Distractor 3 is plausible because the RBM upscale setpoint calculation is analyzed and included in the Core Operating Limits Report that is cycle specific so it could be different, but incorrect because the RBM allowable value formulae are currently the same for Unit 1 and Unit 2.

Reference: ITS 2.0, B2.0, Appendix I, and Appendix J

Reference provided during examination: None

Cognitive level: Memory

10CFR Part 55: 41.5

Importance Level (RO/SRO): 3.8/3.9

Tier: 3 Group: N/A

PRA: No

K/A: 2.2.3 (Multi-unit license) Knowledge of the design, procedural, or operational differences between units

Safety Function: N/A

SRO Justification: N/A

Question Source: New

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Question History:

Comments:

Objective #: 070.00.04

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

67

ID: 2545673

Points: 1.00

Unit 1 is at 100% RTP.

- A normally open valve needs to be closed to isolate a leak
- No procedural direction is given to close the valve
- An Equipment Status Tag (EST) will be used to track the component position

What additional action, if any, is required per OP-AA-108-101, Control of Equipment and System Status?

- A. No additional action is required
- B. Tracked in the notes of a Work Request
- C. Documented in the Issue Report for the leak
- D. Tracked in the Abnormal Component Position Sheet

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Tracked in the Abnormal Component Position Sheet" is the correct answer.

Per OP-AA-108-101

2.4 Equipment Status Tags (EST's): Identify temporary status of equipment position to ensure configuration control. Identify temporary abnormal equipment positioning in conjunction with the Abnormal Component Position Sheet (ACPS). Direct attention to any special condition of an operating system such as outstanding Post Maintenance Test (PMT) or non-preferred equipment use.

Per OP-AA-108-112

2.4. Mispositioned Plant Component:

- 2.4.1. Any active positionable component placed or left out of the required position for existing plant conditions when the component's required position is tracked by one or more of the following status control tools:
- Procedures
 - Tagouts
 - Work Management Process (e.g. work orders)
 - Other similar authorizing document that align or re-align components

Distractor 1 is plausible because EST's are frequently used as info tags for equipment status and is often used by procedures to inform personnel of component position and procedure in progress, but incorrect because this component is being positioned outside of procedural guidance.

Distractor 2 is plausible because approved work management procedures can be used to track the status of component positions, but incorrect because an approved Work Order is used to control component position and Work Orders follow Work Requests in the work management process.

Distractor 3 is plausible because an Issue Report will record the degraded plant conditions and contains a section for "actions taken", but incorrect because Issue Reports are not an approved method for tracking component position.

Reference: OP-AA-108-112, Rev 13; OP-AA-108-105, Rev 12; MA-AA-716-100, Rev 14; OP-AA-109-101, Rev 19; OP-AA-108-101, Rev 19

Reference provided during examination: None

Cognitive level: Memory
10CFR Part 55: 41.10

Importance Level (RO/SRO): 3.9

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Tier: 3 Group:

PRA: No

K/A: G2.2.14 - EQUIPMENT CONTROL Knowledge of the process for controlling equipment configuration or status

Safety Function:

SRO Justification: N/A

Question Source: New

Question History:

Comments:

Objective #: 798.01.01

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

68

ID: 2553950

Points: 1.00

LGA-009, RADIOACTIVITY RELEASE CONTROL, directs operation of ventilation in areas that may be releasing radioactivity including the Turbine Building, Aux Building, Radwaste Building, or Offgas Building.

What is one reason for this action?

- A. Results in recirculation of the Turbine Building atmosphere.
- B. Operating HVAC preserves building accessibility via discharge through an elevated release.
- C. Results in positive pressure inside the Turbine Building to limit the intrusion of radioactivity from the Reactor Building.
- D. Assures that any radioactivity in the Turbine Building is being discharged through a ground level release point to limit the dispersion of the radioactivity.

Answer: B

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "Operating HVAC preserves building accessibility." Is the correct answer.

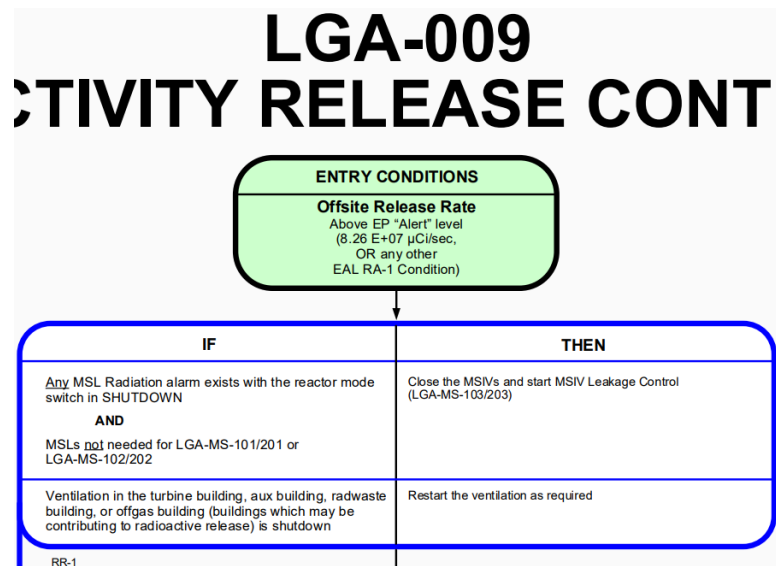
Operating ventilation prevents a ground level release, preserves building accessibility and discharges radioactivity through an elevated, monitored release point.

Per LPGP-PSTG-02C LGA BASES VOLUME II, LGAS FOR HOT CONDITIONS, page 237 RR-1

RR-1 OR#2

Continued personnel access to the turbine building, auxiliary building, radwaste building, or offgas buildings outside the secondary containment boundary may be essential for responding to emergencies or transients which may degrade into emergencies. These buildings are not always airtight structures, and radioactivity release inside the buildings would not only limit personnel access, but would eventually lead to an unmonitored ground level release. Operating HVAC preserves building accessibility and discharges radioactivity through an elevated, monitored release point.

Per LGA-009, RADIOACTIVITY RELEASE CONTROL (1-3)



Distractor 1, 2 and 3 are all plausible only because each mentions the Turbine Building in the response.

Reference: LGA-009, RADIOACTIVITY RELEASE CONTROL (1-3), rev 9

Lesson Plan 509, Section IV.B., rev 10

UFSAR chapter 15.6.4

LPGP-PSTG-02C, LGA BASES VOLUME II, LGAS FOR HOT CONDITIONS, rev 0

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Reference provided during examination: none

Cognitive Level: Memory
10CFR Part 55: 41.(b)(11)

Importance Level (RO/SRO): 3.8
Tier: 3 Group:

PRA: No

K/A: G 2.3.11 RADIATION CONTROL Ability to control radiation releases

Safety Function:

SRO Justification: N/A

Question Source: Bank

Question History: LaSalle 21-1 NRC ILE

Comments:

Objective: 509.00.03 DISCUSS the LGA-009 RADIOACTIVITY RELEASE CONTROL (1-3) OVERRIDES.

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

69

ID: 2545572

Points: 1.00

Unit 2 is at 100% RTP.

Multiple alarms occur simultaneously.

Priority response will go to the alarm associated with...

- A. a single train of safety equipment ONLY.
- B. Initiating Conditions for an Emergency Action Level (EAL) ONLY.
- C. the first event received on the Sequence of Events Recorder (SER) ONLY.
- D. Immediate Operator Actions of an Abnormal Operating procedure (LOA) ONLY.

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer explanation: "Immediate Operator Actions of an Abnormal Operating procedure (LOA) ONLY." is the correct answer.

Per OP-AA-103-102-1001 Actions to stabilize the plant take priority during transient response. Immediate operator actions are actions that are urgently needed to mitigate a transient, are committed to memory by all licensed operators, and are identified as immediate actions in site specific emergency, off-normal, abnormal, or annunciator response procedures.

Distractor 1 is plausible because Safety systems loss may potentially alter the crew response to a transient, but incorrect because transients are designed with the loss of a single train as part of the design criteria.

Distractor 2 is plausible because Emergency action levels require a prompt response due to indicating a potential to release radioactive material off site, but incorrect because actions to stabilize the plant take initial priority.

Distractor 3 is plausible because when multiple alarms are received the SER is referenced to determine the initiating event, but incorrect because actions to stabilize the plant will be taken prior to determining the initiating event.

Reference: OP-AA-103-102-1001 rev 3; OP-AA-103-102 rev 21

Reference provided during examination: None

Cognitive level: Low

10CFR Part 55: 41.10

Importance Level (RO/SRO): 4.2/4.1

Tier: 3 Group:

PRA: No

K/A: G2.4.31 Knowledge of annunciator alarms, indications, or response procedures

Safety Function: N/A

SRO Justification: N/A

Question Source: New

Question History:

Comments: Objective #: LAS-LO-638-TO

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

70

ID: 2504681

Points: 1.00

Shutdown margin for Units 1 and 2 is the amount of reactivity by which a xenon-free reactor at 68°F would be subcritical if all control rods were...

- A. withdrawn, assuming an average worth rod remains fully inserted.
- B. inserted, assuming an average worth rod remains fully withdrawn.
- C. withdrawn, assuming the highest worth rod remains fully inserted.
- D. inserted, assuming the highest worth rod remains fully withdrawn.

Answer: D

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation: "inserted, assuming the highest worth rod remains fully withdrawn." is the correct answer.

Shutdown margin is defined per TS Definitions:

1.1 Definitions (continued)

SHUTDOWN MARGIN (SDM)

SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that:

- The reactor is xenon free;
- The moderator temperature is $\geq 68^{\circ}\text{F}$, corresponding to the most reactive state;
- All control rods are fully inserted except the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

Distractor 1 is plausible because SDM is defined with a control rod remaining full out, but incorrect because shutdown margin assumes all rods are inserted with only one rod remaining full out

Distractor 2 is plausible because SDM is defined by all rods inserted with the exception of one, but incorrect because the single rod out is assumed to be the highest worth control rod.

Distractor 3 is plausible because SDM is defined with a control rod withdrawn, but incorrect because only one control rod remains withdrawn with all other control rods fully inserted.

Reference: GFES Reactor Theory Chapter 8 REACTOR OPERATIONAL PHYSICS, TS Definitions for SDM

Reference provided during examination: None

Cognitive level: Memory
10CFR Part 55: 41(b)(1)

Importance Level (RO/SRO): 3.5
Tier: 4 Group:

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

PRA: No

K/A: 292002 K1.10 Neutron Life Cycle Describe the neutron life cycle using the following terms: Define shutdown margin

Safety Function: 1

SRO Justification: N/A

Question Source: Bank

Question History:

Comments:

Objective #: Define shutdown margin.

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

71

ID: 2545569

Points: 1.00

Unit 2 has been at 100% RTP for two weeks.

- Reactor power was lowered to 60% over 1 hour
- The load drop will last an additional 4 hours

To maintain reactor power at 60% RTP, the operator must add _____, because the xenon-135 concentration is _____.

- A. negative reactivity; rising
- B. negative reactivity; lowering
- C. positive reactivity; rising
- D. positive reactivity; lowering

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer explanation: "positive reactivity; rising" is the correct answer.

Per LGP-3-1 Attachment D, equilibrium xenon concentration is achieved in about 40 hours for a typical BWR. Since Unit 2 has been operating for 2 weeks, or 336 hours, the xenon concentration prior to the load drop was at 100%. Following the power reduction to 50% the production and removal rate of xenon-135 change with the power reduction and the removal rate due to neutron flux is the dominant factor. Thus less xenon is being removed by the neutron flux than at 100% RTP but the production rate of xenon from the decay of iodine-135 is based on the half-life of iodine-135 and the concentration of iodine-135 with its associated production of xenon-135 will respond more slowly.

Per LGP-3-1 xenon concentration is expected to peak between 8-10 hours following a load reduction from 100% RTP with 100% equilibrium xenon concentration. Since the total time for the load reduction will be 5 hours, xenon-135 concentration will be rising the entire time.

Since xenon-135 adds negative reactivity to the core, the reactor operator will need to add positive reactivity.

Distractor 1 is plausible because rising concentrations of other isotopes, such as plutonium, add positive reactivity to the core, but incorrect because xenon-135 is a poison and lowers core reactivity as the concentration rises.

Distractor 2 is plausible because the operator will add positive reactivity to compensate for the xenon-135 concentration change, but incorrect because xenon concentration will be rising following the power reduction.

Distractor 3 is plausible because negative reactivity needs to be added to compensate for lowering xenon-135 concentration, but incorrect because xenon-135 concentration will be rising over the specified time period due to the power reduction.

Reference: LGP-3-1 rev 76; GFES Chapter 6 Fission Product Poisons Instructor Guide, rev 4

Reference provided during examination: None

Cognitive level: High
10CFR Part 55: 41.1

Importance Level (RO/SRO): 3.2
Tier: 4 Group: N/A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

PRA: No

K/A: 292006 K1.14 REACTOR THEORY: Explain the process and reason for the reactor operator to compensate for the time dependent behavior of xenon-135 concentration in the reactor

Safety Function: 1 Reactivity Control

SRO Justification: N/A

Question Source: Bank

Question History:

Comments:

Objective #: 303.00.01

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

72

ID: 2545559

Points: 1.00

Unit 1 is at 100% RTP.

1A Reactor Recirculation pump trips.

Reactor power will initially _____ due to the effects of the _____ coefficient.

- A. rise; void
- B. lower; void
- C. rise; moderator temperature
- D. lower; moderator temperature

Answer: B

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer explanation: "lower; void" is the correct answer.

When a sudden reduction in coolant flow rate occurs from the loss of a recirculation pump, less voids are swept through the core and power lowers due to increase in the void fraction in the core.

Distractor 1 is plausible because power can initially rise during evolutions to lower core reactivity due to void fraction changes, but incorrect because these changes are associated with control rod movements not core flow changes.

Distractor 2 is plausible because core pressure and saturation temperature will lower due to the power reduction and with a negative moderator coefficient will add positive reactivity, but incorrect because reactor power will initially lower due to the change in mass flow rate and the increase if core void fraction.

Distractor 3 is plausible because reactor power will lower, but incorrect because the initial power reduction is from the change in mass flow rate and the increase if core void fraction.

Reference: GFES Chapter 8 REACTOR OPERATIONAL PHYSICS Instructor Guide Rev 4

Reference provided during examination: None

Cognitive level: Memory

10CFR Part 55: 41.1

Importance Level (RO/SRO): 3.7

Tier: 4 Group: N/A

PRA: No

K/A: 292008 K1.26 REACTOR THEORY: Explain reactor power response to decrease in core flow

Safety Function: 1 Reactivity Control

SRO Justification: N/A

Question Source: Bank

Question History:

Comments:

Objective #: 022.00.01

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

73

ID: 2545557

Points: 1.00

Unit 2 is at 100% RTP.

Steam to 23A Feedwater heater isolates.

Feedwater...

- A. quality will rise.
- B. density will lower.
- C. enthalpy will lower.
- D. temperature will rise.

Answer: C

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer explanation: "enthalpy will lower." is the correct answer.

Per UFSAR section 10.4.7.1.2, feedwater temperature entering the RPV is 425.1°F. At normal RPV operating pressure, 1003 psig, saturation temperature is approximately 546°F, therefore feedwater is a subcooled liquid. Steam is used to add heat to feedwater prior to entering the RPV. With the loss of steam to a feedwater heater this is equivalent to removing heat from the final feedwater entering the RPV. As heat is removed from a subcooled liquid the enthalpy will lower.

Distractor 1 is plausible because the steam no longer cooled by the feedwater heater will have a higher quality, but incorrect because feedwater is currently subcooled.

Distractor 2 is plausible because rising temperature causes density to lower, but incorrect because temperature will lower causing density of the subcooled liquid to rise.

Distractor 3 is plausible because temperature changes, but incorrect because temperature will lower.

Reference: UFSAR chapter 10.4; ASME Steam Tables

Reference provided during examination: None

Cognitive level: High

10CFR Part 55: 41.14

Importance Level (RO/SRO): 2.8

Tier: 4 Group: N/A

PRA: No

K/A: 293003 K1.16 THERMODYNAMICS THEORY: STEAM: Define the following term: SUBCOOLING

Safety Function: 4 Heat Removal From the Reactor Core

SRO Justification: N/A

Question Source: Bank

Question History:

Comments:

Objective #: 77.00.16 (293003)

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

74

ID: 2545553

Points: 1.00

Unit 1 is at 100% RTP.

- The online Steam Jet Air Ejector (SJAE) is inadvertently isolated from the Main Condenser
- Condenser cooling water parameters remain constant

If NO operator action is taken over the next 60 minutes, condenser backpressure will...

- A. remain the same.
- B. DEGRADE until at atmospheric pressure.
- C. DEGRADE and stabilize still at a vacuum.
- D. IMPROVE and stabilize still at a vacuum.

Answer: B

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer explanation: " slowly RISE until at atmospheric pressure." is the correct answer.

With the suction from the online steam jet air ejector isolated, non-condensable gasses are no longer being evacuated from the main condenser and will build up in the main condenser air/steam space. As the non-condensable gasses build up they cover the heat exchange surface of the condenser preventing the cooldown of the steam exiting the main turbine. With no changes to cooling water flow parameters the loss of the condensing action in the main condenser cause pressure to continuously rise toward atmospheric pressure.

Per LOA-OG-101 the amount of time condenser vacuum will be maintained without the off gas system will depend on air-leakage into the main condenser and power reduction will raise the time available to recover off gas from a few minutes up to an hour.

Distractor 1 is plausible because condenser cooling water system parameters have not changed, but incorrect because with a loss of the SJAE removing non-condensable gases the condenser pressure will rise toward atmospheric pressure.

Distractor 2 is plausible because Off-gas systems are used to draw initial vacuum in the main condenser without the condensing action of steam, but incorrect because the steam jet air ejectors remove non-condensable gases from the main condenser to prevent them from impeding the condensing action of steam and without those gases being removed the condenser pressure will continue to rise.

Distractor 3 is plausible because vacuum is expected to lower and the concept of vacuum, the pressure below atmospheric pressure, and absolute pressure in the main condenser are often confused, but incorrect because absolute pressure in the main condenser is expected to rise.

Reference: LOA-OG-101 rev 33, Generic Physics Corp BWR GFES Chapter 4 Thermodynamic Processes Instructor guide rev 4

Reference provided during examination: None

Cognitive level: High
10CFR Part 55: 41.14

Importance Level (RO/SRO): 2.6
Tier: 4 Group: N/A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

PRA: No

K/A: 293004 K1.13 THERMODYNAMIC PROCESSES Condensers: Explain vacuum formation in condenser processes.

Safety Function: 2 Reactor Water Inventory Control

SRO Justification: N/A

Question Source: Bank

Question History:

Comments:

Objective #: 75.00.16 (293004)

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

75

ID: 2545346

Points: 1.00

Unit 1 was at 100% RTP when an ATWS occurred.

Standby Liquid pump 1A is injecting with the following initial parameters:

- RPV pressure = 1000 psig
- Flow rate = 42 gpm

If RPV pressure lowers to 500 psig, SBLC pump flow rate will...

- A. remain constant
- B. rise in direct proportion to the change in pump differential pressure
- C. rise in direct proportion to the square of the change in pump differential pressure
- D. rise in direct proportion to the square root of the change in pump differential pressure

Answer: A

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864

Answer Explanation

Answer explanation: "remain constant" is the correct answer.

Standby liquid pumps are positive displacement pumps. Positive displacement pumps are designed to inject at a given flow rate for any pressure. The Standby liquid injection system is designed to inject at 42 gpm and is periodically tested at 1220 to 1230 psig.

Distractor 1 is plausible because this is the flow characteristics expected through a hole or an orifice, but incorrect because positive displacement pumps will provide a constant flow regardless of differential pressure across the pump.

Distractor 2 is plausible because this is a pump law for changes in pump speed, but incorrect because pump speed will not change based on changes in system backpressure.

Distractor 3 is plausible because this is a pump law associated with changes in pump speed, but incorrect because positive displacement pump speed will remain constant regardless of system backpressure.

Reference: LOS-SC-R5 rev 15

Reference provided during examination: None

Cognitive level: High

10CFR Part 55: 41.14

Importance Level (RO/SRO): 2.7

Tier: 4 Group: N/A

PRA: Yes

K/A: 293006 K1.30 Describe the methods of controlling system flow rates

Safety Function: 1 Reactivity Control

SRO Justification: N/A

Question Source: Bank

Question History:

Comments:

Objective #: 28.00.11

Test Answer Key

LaSalle ILT 23-1 NRC RO

Test ID: 375864
