Limerick Op-Test Change Summary

JPM B

Changed alternate path initiator

Admin JPM A-3

Replaced JPM

Admin JPM A-4

Added additional complexity to offsite release JPM

Scenario #3

- Deleted CT-1
- Deleted Event 1

Note:

The following 2015 Limerick written exam has the subheading of "LGS 2015 ILT NRC EXAM – **SRO**." This SRO written exam is a 100 question exam, where the first 75 questions are the same 75 questions which comprise the entire 2015 Limerick RO written exam, and are in the same sequential order as the questions on the RO exam. For this reason, the RO written exam is not separately included.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

1 **ID: 1098269** **Points: 1.00**

WHICH ONE of the following identifies the location of the breaker that supplies electrical power directly to the Unit 2 Instrument Air Compressor, 2BK101?

A. 224A

B. 224A-G-F

C. D244

D. D244-R-H

Answer: D

Answer Explanation

Refer to 2S15.1.A (COL-2), 2B Instrument Air Compressor Normal Operating Lineup. Page 7 of 7, Step 62 shows Div 4, 480 volt MCC D244-R-H supplying the 2BK101 compressor. Note - Although powered from a Divisional MCC, the compressor is shunt-tripped on a LOCA condition, meaning that the compressor will be load-shed and will not be available until operators "reset" the shunt trip device, then allowing a restart of the compressor (refer to SE-10-1, Breaker Reset Following a LOCA for more detail).

'D' is correct: D244-R-H. Correct for the reason described above.

'A' is wrong: 224A. 224A is a non-safeguard 480 volt Load Center (LCC). This choice is plausible to the Examinee who incorrectly believes that the 2BK101 air compressor has a motor large enough as to be powered from an LCC and, once that Examinee has believed that, he/she then recalls that the Instrument Air Compressors are, themselves, non-safeguard qualified components (a true fact); that Examinee forgets, however, that the compresors are nonetheless Divisionally powered, as described above.

'B' is wrong: 224A-G-F. This is an MCC that is fed from the 224A LCC already described in choice 'A'. Plausible to the Examinee who does recognize that the air compressor motor is small enough to be pwered from an MCC, but who nonetheless believes that the MCC is a non-safeguard one for the same reason as already described for choice 'A'.

'C' is wrong: D244. As already described, this is the LCC that powers MCC D244-R-H, but it is not the "location" of the circuit breaker. Plausible to the examine who correctly recalls that the 2BK101 compressor is a Div 4 load, but who incorrectly recalls the size of its motor (i.e., believing it to be large enough so as to be powered directly from the LCC).

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 1 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.00	
System ID:	1098269	
User-Defined ID:	REV 02, 01/06/15	
Cross Reference Number:		
Topic:	Recall power supply to Instrument Air Compressor 2BK101	
Num Field 1:	2.8	
Num Field 2:	2.8	
Text Field:	300000 K2.01	
Comments:	Level	RO
	Tier	2
	Group	1
	KA # and Rating	300000 K2.01 (2.8/2.8)
	KA Statement	300000 Instrument Air Knowledge of electrical power supplies to the following: K2.01 Instrument Air Compressor
	References	2S15.1.A (COL-1), Rev.5 2S15.1.A (COL-2), Rev.4 2S15.1.A (COL-4), Rev.6
	Examinee References	None
	Learning Objective	LGSOPS0015, no specified objective
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.7
	Comments	Major revision to address NRC Rev.00 comments. Added a reference (2S15.1.A (COL-4)) to support revision. Major revision, again (Rev.02), to address concerns identified during Validation of Rev.01.

2 **ID: 1098270** **Points: 1.00**

Unit 1 plant conditions:

- LOCA in progress
- RPV pressure is 380 psig
- RPV level is -80", lowering
- No SRVs are open
- Operators have manually initiated the 1A Core Spray Loop
- A and C Core Spray Pumps are running with suctions from the Suppression Pool
- No other high or low pressure injection systems are running

WHICH ONE of the following describes the status of the Suppression Pool water inventory?

(Do **NOT** consider the steam in the drywell that enters the suppression pool via the downcomers.)

A. Remaining steady

B. Lowering at 6350 gpm

C. Lowering at 3175 gpm

D. Lowering at 775 gpm

Answer: A

Answer Explanation

The capacity of each Core Spray Pump is 3175 gpm. The 1A Core Spray Loop Min Flow Valve (HV-052-1F031A) automatically opens when a pump is running with that Loop's flow less than 775 gpm. When open, pump discharge is directed back to the Suppression Pool. The 1A Core Spray Loop Inboard Injection Valve (HV-052-1F005) is normally-CLOSED and automatically opens with the combination of a Core Spray initiation signal (auto or manual) AND RPV pressure below 455 psig. [S52.7.A, Manual Initiation After Failure of Automatic Injection During a LOCA, can be used to verify the 775 gpm min flow rate as well as the 455 psig automatic opening setpoint for the Inboard Injection Valve.]

'A' is correct: Remaining steady. Stem conditions reveal that RPV pressure is 380 psig; although the open-permissive is satisfied for the 1F005 valve (<455 psig), pressure is still well above the approximate 330 psig shutoff head for these pumps. Therefore, the pumps are not injecting into the RPV but are instead running on min flow (i.e., recircing from/to the suppression pool at a rate of about 775 gpm); thus the pool water inventory is steady. Stem conditions state there are no other sources of water to or from the pool (no SRVs are open; no other injection systems are running); this ensures that only the operating 1A Core Spray Loop is influencing the pool water inventory. For the same reason, the stem question also directs the examinee to "NOT consider the steam that migrates from the drywell to the suppression pool via the downcomers".

'B' is wrong: Lowering at 6350 gpm. This is plausible to the examinee who fails to recognize that, with RPV pressure at 380 psig, the pumps cannot be injecting. That examinee, therefore, selects this choice based on recalling that the combined capacity of the two running pumps (3175 gpm each) is about 6350 gpm. Since the pumps are taking (suctions) from the suppression pool at a rate of 6350 gpm, and simply injecting that water into the RPV in an attempt to mitigate the dropping RPV water level, the pool water inventory is being reduced at about 6350 gpm.

'C' is wrong: Lowering at 3175 gpm. This is plausible for reasons nearly identical to those of choice 'B'. In this case, however, the examinee incorrectly recalls the combined capacity of the two running pumps; instead, that examinee applies the capacity of a single pump (3175 gpm).

'D' is wrong: Lowering at 775 gpm. This is plausible to the examinee who recognizes that the pumps cannot be injecting at an RPV pressure of 380 psig and recalls the min flow valve opening setpoint of 775 gpm (with an initiation signal). However, the examinee's thinking is flawed in that he/she fails to remember that as the pumps (together) are removing water from the pool at a rate of 775 gpm, so too are they returning (recircing) that same water to the pool at the same rate, thus maintaining a constant inventory of pool water.

Question 2 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.50
System ID:	1098270
User-Defined ID:	REV 02, 01/06/15
Cross Reference Number:	LLOT0350.9
Topic:	Recognize how Core Spray Pump operation impacts Supp Pool water inventory
Num Field 1:	3.4
Num Field 2:	3.4
Text Field:	209001 K1.02
Comments:	Level: RO Tier: 2 Group: 1 KA # and Rating: 209001 K1.02 (3.4/3.4) KA Statement: 209001 Low Pressure Core Spray Knowledge of the physical connections and/or cause-effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following: K1.02 Torus/suppression pool References: S52.7.A, Rev.9 Examinee References: None Learning Objective: LLOT0350.9 Question source: New Question history: None Cognitive level: Higher 10 CFR 55: 41.7 Comments: Minor (editorial) to stem and answer choices to address NRC Rev.00 comments. Minor change to stem to address NRC Rev.01 comments.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

3 **ID: 1098287** **Points: 1.00**

Unit 1 is operating at 100% power, with the following:

- 1A SLC Pump is blocked for motor repairs
- 1B and 1C SLC Pumps are aligned for auto-start

1A SLC Pump repairs are now completed; clearances have been removed.

Per S48.1.A (SLC System Set-up for Normal Operation), operators have completed aligning 1A and 1B SLC Pumps for auto-start.

WHICH ONE of the following is an expected SLC related indication in the MCR?

A. C SLCS PUMP AUTO-START STATUS TROUBLE alarm is SEALED-IN

B. 1C SLC Pump white Pump Status light is EXTINGUISHED

C. 1C SLC Pump amber OVERRIDE light is ILLUMINATED

D. 1C SLC Pump white Squib Valve Continuity light is EXTINGUISHED

Answer: C

Answer Explanation

Refer to S48.1.A, Section 4.8, which provides the direction for removing the 1C SLC Pump from an auto-start lineup and placing it back in standby status (note: the "standby" status of the 1C pump does not imply an auto-start capability). NOTE #1 of step 4.8.1 states that at no time time should all 3 pumps be aligned for automatic injection. Follow the sequence of steps 4.8.4 thru 4.8.11. With the 1C Pump still aligned for auto-start, its control switch is in NORM and its INHIBIT-ENABLE switch is in ENABLE. When the operator places its control switch in STOP, the alarm comes in...C SLCS PUMP AUTO-START STATUS TROUBLE...and the 1C pump amber OVERRIDE lamp is LIT. The operator then places the INHIBIT-ENABLE switch in INHIBIT and the alarm clears. Finally, step 4.8.11 has the operator verify the final indications for the 1C pump in its standby status, including: both the Green and White (pump status) lights are LIT (the white light is simply a Power Available light), and the amber OVERRIDE light is LIT.

'C' is correct: 1C SLC Pump amber OVERRIDE light is ILLUMINATED. Correct for the reasons described above.

'A' is wrong: C SLCS PUMP AUTO-START STATUS TROUBLE alarm is SEALED-IN. Refer to alarm response card ARC-MCR-107, A3, which describes the two conditions that will cause this alarm: C SLC Pump control switch in STOP with the INHIBIT-ENABLE switch in ENABLE, or, C SLC Pump control switch in NORM with the other switch in INHIBIT. As described above, S48.1.A steps 4.8.4 thru 4.8.11 left the 1C pump control switch in STOP with the INHIBIT-ENABLE switch in INHIBIT; therefore, this alarm in not sealed-in. Plausibility of this choice is based on inexperienced control room operators still lacking a solid understanding of the signnificance of this alarm and the particular conditions (not at all intuitive) that cause it.

'B' is wrong: 1C SLC Pump white Pump Status light is EXTINGUISHED. This is plausible to the examinee who does not recall what this WHITE light means with respect to pump status. Often, a LIT white pump status light signifies that the pump is in a Standby status, ready for an automatic start (such as on the trip of the running pump(s)). Because this examinee does recall that the 1C SLC Pump (albeit in "standby" status) nevertheless will not auto-start, and he/she also recalls that the same lights belonging to the 1A and 1B pumps (which are capable of automatically starting) are in fact LIT, the examinee concludes that the white light for the 1C pump must be extinguished.

'D' is wrong: 1C SLC Pump white Squib Valve Continuity light is EXTINGUISHED. This is plausible to the examinee who incorrectly concludes that taking the 1C pump out of an auto-start lineup necessarily removes power from the associated squib valve (i.e., since there is no need for the pump to auto-start, there can be no need for the 1C squib valve to fire).

Question 3 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.50	
System ID:	1098287	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LLOT0048	
Topic:	Recognize MCR indications associated with a normal SLC Pump alignment	
Num Field 1:	4.0	
Num Field 2:	4.1	
Text Field:	211000 A1.09	
Comments:	Level	RO
	Tier	2
	Group	1
	KA # and Rating	211000 A1.09 (4.0/4.1)
	KA Statement	211000 Standby Liquid Control System Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: A1.09 SBLC system lineup
	References	S48.1.A, Rev.22 ARC-MCR-107, A3, Rev.1
	Examinee References	None
	Learning Objective	LLOT0048, no specified objective
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.7
	Comments	Minor editorial revision to address NRC Rev.00 comments.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

4 **ID: 1098291** **Points: 1.00**

Unit 1 plant conditions:

- Reactor scrammed on a total loss of feedwater
- RPV level is -50", down fast
- RPV pressure is 800 psig, steady
- Drywell pressure is 0.4 psig, steady

At T=0:

- 1D RHR Pump starts on a valid -129" RPV level signal
- All other ECCS pumps fail to start

WHICH ONE of the following identifies when the ADS SRVs will automatically OPEN, if at all?

A. Will not automatically open

B. In 105 seconds

C. In 420 seconds

D. In 525 seconds

Answer: D

Answer Explanation

The 1D RHR Pump running provides a permissive to the Div 3 ADS logic. Once RPV level reaches -129", a 420-second timer starts, bypassing the need for a concurrent High Drywell Pressure (1.69 psig) signal. When this 420-second timer expires, it allows the logic to then start the usual 105-second ADS Initiation Timer. When this 105-second timer expires (a total of 525 seconds (420 + 105 = 525) after the -129" level signal was received), the 5 ADS SRVs (S, H, M, E, K) will automatically OPEN. [To validate this information, refer to alarm response cards ARC-MCR-110, B3 and C4.]

'D' is correct: In 525 seconds. Correct for the reasons described above.

'A' is wrong: Will not automatically open. Plausible to the examinee who believe that until there exists the High DW Pressure signal, the ADS Initiation logic will remain as-is; that examinee has forgotten about the High DW Pressure Bypass Timer.

'B' is wrong: In 105 seconds. Plausible to the examinee who does not recognize the absence of the High DW Pressure signal and so believes that the 129" level signal is sufficient to immediately start the ADS Initiation (105-second) Timer.

'C' is wrong: In 420 seconds. Very plausible. A common mistake is for the examinee to forget that the 105-second timer piggy-backs on the 420-second timer; rather, that examinee thinks in terms of the 420 second timer substituting for the 105-second timer...resulting in the auto-opening of the SRVs as soon as the 420-second timer expires.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 4 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.50	
System ID:	1098291	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LGSOPS0050.IL4	
Topic:	Predict when ADS SRVs will automatically open	
Num Field 1:	3.9	
Num Field 2:	3.9	
Text Field:	218000 K1.06	
Comments:	Level	RO
	Tier	2
	Group	1
	KA # and Rating	218000 K1.06 (3.9/3.9)
	KA Statement	218000 ADS Knowledge of the physical connections and/or cause-effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: K1.06 Safety/relief valves
	References	LGSOPS0050, ADS lesson plan
	Examinee References	None
	Learning Objective	LGSOPS0050.IL4
	Question source	Bank 556255
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.7
	Comments	Fixed typo per NRC Rev.00 comments.

5 **ID: 1118907** **Points: 1.00**

Unit 1 reactor startup is in progress.

Per the Control Rod Move Sheet:

- Rod 26-35 is to be withdrawn from position 12 to position 32 (the "**target**" position)
- Continuous Withdraw is permitted
- This is a non-channel distortion rod

Consider the requirements of the following procedures:

- S73.1.A, Normal Operation of RMCS
- OP-AB-300-1001, BWR Control Rod Movement Requirements

WHICH ONE of the following describes a control rod movement that **COMPLIES WITH** the requirements of both of the above listed procedures?

A. The operator single-notches to position 26.
Starting at position 26, the operator uses continuous withdraw; he releases the pushbuttons at position 30 and the rod settles at position 32.

B. The operator single-notches to position 20.
Starting at position 20, the operator uses continuous withdraw; he releases the pushbuttons at position 26 and the rod settles at position 30.

C. The operator uses continuous withdraw; he releases the pushbuttons at position 24 and the rod settles at position 26.
Starting at position 26, the operator uses continuous withdraw; he releases the pushbuttons at position 28 and the rod settles at position 30.

D. The operator uses continuous withdraw; he releases the pushbuttons at position 22 and the rod settles at position 24.
Starting at position 24, the operator uses continuous withdraw; he releases the pushbuttons at position 30 and the rod settles at position 32.

Answer: B

Answer Explanation

Refer to OP-AB-300-1001, the following Steps...

4.6. When using the continuous rod insert or continuous rod withdrawal feature, then the Licensed Operator shall **STOP** control rod movement ***such that the control rod settles at least one notch prior to the target notch position and USE single notch movement to reach the final target rod position.***

4.6.1. The continuous rod insert or continuous rod withdrawal feature shall **not** be used for control rod movements of three notches or less (e.g., moving from position 00 to position 06). **USE** single notch insert or single notch withdrawal to perform control rod movements of three notches or less.

Refer to S73.1.A, the following Steps...

4.3.10 **WHEN** control **rod is located two notches before target position, THEN RELEASE** WITHDRAW **AND** CONTINUOUS WITHDRAW pushbuttons.

4.3.12 **IF** control rod settles short of target position, **THEN PLACE** the control rod in target position using **single notch** withdraw per Section **4.2**.

'B' is correct: The operator single-notches to position 20. Starting at position 20, the operators uses continuous withdraw; he releases the pushbuttons at position 26 and the rod settles at position 30. This move complies with all of the above listed requirements.

'A' is wrong: The operator single-notches to position 26. Starting at position 26, the operator uses continuous withdraw; he releases the pushbuttons at position 30 and the rod settles at position 32. Starting from position 26, the operator was only three notches short of the target of position 32; by applying a continuous withdraw signal, he violated Step 4.6.1 of OP-AB-300-1001. He also violated Step 4.3.10 of S73.1.A by not releasing the pushbuttons until position 30 (just one notch before the target position of 32).

'C' is wrong: The operator uses continuous withdraw; he releases the pushbuttons at position 24 and the rod settles at position 26. Starting at position 26, the operator uses continuous withdraw; he releases the pushbuttons at position 28 and the rod settles at position 30. Starting from position 26, the operator was only three notches short of the target of position 32; by applying a continuous withdraw signal, he violated Step 4.6.1 of OP-AB-300-1001.

'D' is wrong: The operator uses continuous withdraw; he releases the pushbuttons at position 22 and the rod settles at position 24. Starting at position 24, the operator uses continuous withdraw; he releases the pushbuttons at position 30 and the rod settles at position 32. Although the operator did comply with Step 4.6.1 of OP-AB-300-1001 (intending to target position 32 starting from position 24 (4 notches away), he nonetheless violated Step 4.3.10 of S73.1.A by not releasing the pushbuttons until position 30 (just one notch before the target position of 32).

Because of the complexity of how each of the procedural requirements are interrelated, all of the distractors are plauslble when presented in a from-memory format.

Question 5 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	1118907
User-Defined ID:	REV 02, 01/06/15
Cross Reference Number:	LGSOPS2010.13C
Topic:	RMCS - Recall Requirements for Continuous withdraw and single-notch rod moves
Num Field 1:	4.3
Num Field 2:	4.4
Text Field:	201002 2.1.23
Comments:	Level RO

	Tier	2
	Group	2
	KA # and Rating	201002 2.1.23 (4.3/4/4)
	KA Statement	201002 RMCS 2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation.
	References	S73.1.A, Rev.50 OP-AB-300-1001, Rev.07
	Examinee References	None
	Learning Objective	LGSOPS2010.13C
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.10
	Comments	Replacement question (and necessary KA swap to support) to correct problem of overlap with Operating Test.

6 **ID: 1098294** **Points: 1.00**

Unit 1 PRO is preparing to parallel D12 DG with the 201 Safeguard Bus, per ST-6-092-312-1 (D12 DG Slow Start Operability Test Run).

WHICH ONE of the following describes how the PRO prevents a D12 DG trip on reverse power, per the ST?

As soon the D12 DG output breaker closes, the PRO is directed to immediately load the DG to...

A. 100-150 KW.

B. 200-300 KW.

C. 100-150 KVAR.

D. 200-300 KVAR.

Answer: B

Answer Explanation

Refer to ST-6-092-312-1, steps 4.8.10 thru 4.8.12. Step 4.8.10 closes the DG output breaker. Step 4.8.11 directs the operators to immediately load the DG to 200-300 KW using the Speed Governor. Step 4.8.12 directs the operator to immediately load the DG to 100-150 KVAR.

'B' is correct: 200-300 KW. Per step 4.8.11, a real load of 200-300 KW must be immediately picked up by the DG in order to prevent actuating the "reverse power" relay (a device that senses ONLY "real" power (KW); it does not sense/respond to "reactive" power (KVAR)).

'A' is wrong: 100-150 KW. Plausible to the examinee who recalls a "100-150" band, but who confuses KVAR loading requirement as being associated with preventing a reverse power trip.

'C' is wrong: 100-150 KVAR. Plausible to the examinee who probably recalls both values from the ST (real load value and reactive load value), but who incorrectly associates the KVAR requirement with a reverse power trip. Especially plausible because the purpose of the step to immediately pick up the 100-150 KVAR is to ensure that DG remains in a "vars-out" condition and does not become a "vars-in" machine...a condition for which the DG is unanalyzed.

'D' is wrong: 200-300 KVAR. Plausible to the examinee who correctly recalls a "200-300" value but who confuses KVAR loading requirement as being associated with preventing a reverse power trip.

EXAMINATION ANSWER KEY

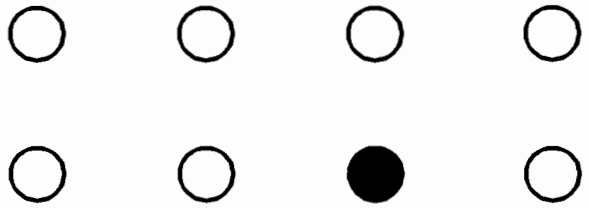
LGS 2015 ILT NRC EXAM - SRO

Question 6 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	1098294
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LGSOPS0092B.IL6
Topic:	Recall how the operator prevents a reverse power D/G trip when paralleling
Num Field 1:	3.0
Num Field 2:	3.1
Text Field:	264000 A1.09
Comments:	Level RO Tier 2 Group 1 KA # and Rating 264000 A1.09 (3.0/3.1) KA Statement 264000 EDGs Ability to predict and/or monitor changes in parameters associated with operating the EMERGENCY GENERATORS (DIESEL/JET) controls including: A1.09 Maintaining minimum load on emergency generator (to prevent reverse power) References ST-6-092-312-1, Rev.96 Examinee References None Learning Objective LGSOPS0092B.IL6 Question source New Question history None Cognitive level Lower 10 CFR 55 41.5 Comments None

7 **ID: 1098325** **Points: 1.00**

Refer to the figure below to answer this question.

SCRAM SYSTEM LOGIC



Unit 1 is operating at 100% power when the following occurs:

- One of the SCRAM SYSTEM LOGIC (white) lights extinguishes at10C603
- The bulb is good

WHICH ONE of the following correctly interprets this indication?

A. One RPS 'A' Channel is tripped.

B. One RPS 'B' Channel is tripped.

C. One-quarter (1/4) of the scram solenoid pilot valves have re-positioned.

D. One-quarter (1/4) of the control rods have a de-energized solenoid.

Answer: D

Answer Explanation

A look at these 8 lights (at 10C603) will remind the examinee that the 4 lights across the top row (left-to-right) are labeled A1, A2, A3, A4, representing 120 VAC RPS power being supplied to the 'A' solenoid for the control rod HCUs in each of 4 rod Groups (Groups 1, 2, ,3, 4, left-to-right). Similarly, the bottom row of lights are labeled B1, B2, B3, B4, and monitor the 120 VAC RPS power supplied to the 'B' solenoid for each rod Group. The extinguished 'B3' light (in the case of this question) indicates that RPS power is no longer being supplied to the 'B' solenoids for one of 4 Groups of rods; i.e., one-quarter of the control rods have a de-energized solenoid (the 'B' solenoid in this case). Such an indication can only occur as a result of a malfunction, most likely a blown fuse in the circuit that supplies 120 VAC RPS power to the 'B' solenoids of that one rod Group.

'D' is correct: One-quarter (1/4) of the control rods have a de-energized solenoid. Correct for the reasons described above.

'A' is wrong: One RPS 'A' Channel is tripped. If a single RPS 'A' Channel (for example: Channel A1) were to trip, all 4 lights across the TOP row would extinguish; i.e., the indication expected for an RPS 'A' half-scram. Plausible to the examinee who either, too quickly jumps at the idea that a half-scram is indicated here, or who does not completely understand what each of these 8 lights represents.

'B' is wrong: One RPS 'B' Channel is tripped. If a single RPS 'B' Channel (for example: Channel B2) were to trip, all 4 lights across the BOTTOM row would extinguish; i.e., the indication expected for an RPS 'B' half-scram. Plausible for the same reason as for choice 'A'.

'C' is wrong: One-quarter (1/4) of the scram solenoid pilot valves have re-positioned. As a result of the ASCO solenoid valve Mod (ECR 04-00185), each control rod HCU only has one, dual-solenoid scram pilot solenoid valve. "Re-positioning" a pilot valve would take air off of the actuators for the Scram Inlet and Scram Outlet valves, causing that rod to scram into the core. The only way to re-position the pilot is by de-energizing both of its solenoids ('A' and 'B'). Before the ASCO Mod, each HCU had two single-solenoid scram pilot solenoid valves; the solenoid for one of the valves was powered by RPS 'A', the other valve's solenoid was powered by RPS 'B'. If power were interrupted to the solenoid for the RPS 'B' valve (for example), that one valve would re-position; however, the associated rod would not scram because the valve powered by RPS 'A' would be unaffected and not re-position, i.e., air would still be supplied to the HCU. This choice is plausible to the examinee confuses the "before Mod & after Mod" configurations for these valves.

Question 7 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	2.50	
System ID:	1098325	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	LLOT0071.9D	
Topic:	Interpret extinguished Scram System Logic (white) lights	
Num Field 1:	3.4	
Num Field 2:	3.3	
Text Field:	212000 A1.11	
Comments:	Level	RO
	Tier	2
	Group	1
	KA # and Rating	212000 K3.06 (4.0/4.1)
	KA Statement	212000 Reactor Protection System Knowledge of the effect that a loss or malfunction of the REACTOR PROTECTION SYSTEM will have on following: K3.06 Scram air header solenoid operated valves
	References	C71-1020-E-009, Rev.16 C71-1020-E-010, Rev.28
	Examinee References	None
	Learning Objective	LLOT0071.9D
	Question source	Bank 894672

	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.7
	Comments	None

8 **ID: 1098327** **Points: 1.00**

Plant conditions:

- Unit 1 is in OPCON 4 with drywell shield block removal in progress
- Unit 2 is in OPCON 1

Unit 2 drywell pressure rises to 2.1 psig.

Consider the following Secondary Containment Ventilation Zones:

- Zone 1
- Zone 2
- Zone 3

WHICH ONE of the following identifies (from the above list) **ALL** of the the Secondary Containment Ventilation Zones that have aligned to SGTS?

A. Zone 2

B. Zone 2
Zone 3

C. Zone 3

D. Zone 1
Zone 3

Answer: A

Answer Explanation

During normal plant operations, ventilation is provided to three major Zones of the Secondary Containment; they are: Zone 1 (Unit 1 Reactor Enclosure), Zone 2 (Unit 2 Reactor Enclosure), and Zone 3 (the Common Refuel Floor). When one of the Units is shutting down for a refueling outage, prior to reaching OPCON 4, that Unit's Zone is inter-tied (interlocked) with Zone 2, to accomodate the removal of the drywell shield blocks. While the Zones are inter-tied, an isolation signal on one of those Zones will cause the other Zone to isolate, as well. For example: a Zone 1 isolation signal will isolate Zone 3, as well, and vice versa. Isolated Zones are automatically aligned to SGTS.

'A' is correct: Zone 2. With Unit 2 in OPCON 1, its ventilation Zone (Zone 2) is not inter-tied with any other. Therefore, a Unit 2 high drywell pressure signal will isolate only Zone 2.

'B' is wrong: Zone 2, Zone 3. Plausible to the examinee who confuses the physical arrangement of the three Zones, thinking that the they are designated 1-2-3 according to the left-to-right arrangement of the two Units with the common refuel floor in between (1-refuel floor-3); that examinee also belives that Zones 2 and 3 are inter-tied, thus isolating both of those Zones in response to the high DW pressure signal.

'C' is wrong: Zone 3. Plausible to the examinee who knows that only the Unit 2 ventilation Zone will isolate, but confuses the physical arrangement of the three Zones, thinking that the they are designated 1-2-3 according to the left-to-right arrangement of the two Units with the common refuel floor in between (1-refuel floor-3).

'D' is wrong: Zone 1, Zone 3. Plausible to the examinee who recognizes that two Zones are currently inter-tied, but who mistakenly relates Zone 1 to Unit 2, rather than Unit 1.

Question 8 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.50	
System ID:	1098327	
User-Defined ID:	REV 02, 01/06/14	
Cross Reference Number:	LLOT0076B.6	
Topic:	Predict Zones that isolate in response to SGTS initiation signal	
Num Field 1:	3.1	
Num Field 2:	3.3	
Text Field:	223002 K1.12	
Comments:	Level	RO
	Tier	2
	Group	1
	KA # and Rating	223002 K1.12 (3.1/3.3)
	KA Statement	223002 PCIS/NSSSS Knowledge of the physical connections and/or causeeffect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF and the following: K.12 Standby gas treatment system
	References	LLOT0076A, Reactor Enclosure HVAC lesson plan LLOT0076B, Secondary Containment lesson plan
	Examinee References	None
	Learning Objective	LLOT0076A.3
	Question source	Bank 560639 *
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.7
	Comments	*Minor revision to address NRC Rev.00 comments. Editorial to stem and answer choices to address NRC Rev.01 comments.

9 **ID: 1102907** **Points: 1.00**

An EQUALIZE charge is in progress on the Div 1 125 VDC Battery.

The procedure requires the charger to be placed back in FLOAT mode when the charging current drops to 5 amps.

WHICH ONE of the following describes a potential consequence of leaving the charger in EQUALIZE mode until the charging current drops to 3 amps?

A. Charging current may exceed the Tech Spec limit of 3 amps when the charger is placed back in FLOAT mode

B. Charging current may exceed the Tech Spec limit of 1 amp when the charger is placed back in FLOAT mode

C. The two chargers for the bus may be unevenly loaded when the charger in placed back in FLOAT mode

D. Excessive hydrogen may be generated by the battery until the charger is placed back in FLOAT mode

Answer: D

Answer Explanation

The discussion below is an excerpt from the D.O.E. Fundamentals Handbook, Electrical Science, Volume 2 of 4; specifically: Module 4, "Batteries".

A lead-acid battery cannot absorb all the energy from the charging source when the battery is nearing the completion of the charge. This excess energy dissociates water by way of electrolysis into hydrogen and oxygen. Oxygen is produced by the positive plate, and hydrogen is produced by the negative plate. This process is known as gassing. ***Gassing is first noticed when cell voltage reaches 2.30-2.35 volts per cell and increases as the charge progresses.*** At full charge, the amount of hydrogen produced is about one cubic foot per cell for each 63 ampere-hours input. If gassing occurs and the gases are allowed to collect, an explosive mixture of hydrogen and oxygen can be readily produced. It is necessary, therefore, to ensure that the area is well ventilated and that it remains free of any open flames or sparkproducing equipment.

'D' is correct: Excessive hydrogen may be generated by the battery until the charger is placed back in FLOAT mode. Correct for the reasons described above.

'A' is wrong: Charging current may exceed the Tech Spec limit of 3 amps when the charger is placed back in FLOAT mode. By leaving the charger in Equalize mode until the charging current drops another 2 amps (i.e., from 5 amps down to 3 amps, as suggested by this question), there is no potential for damaging the battery due to "overcharging" it. Therefore, when the charger is placed back in FLOAT mode, there is no expectation that the FLOAT current will be higher than normal. Even if the Examinee were to believe that the small bit of overcharging could damage the battery, causing it to draw an excessive amount of current when the charger is placed back in FLOAT mode, this answer choice is nonetheless wrong because it suggests that the Tech Spec limit for this Div 1 Battery is "less than 3 amps"; it is not...the limit is 2 amps (refer to Tech Spec SR 4.8.2.1.a.1 for this value).

'B' is wrong: Charging current may exceed the Tech Spec limit of 1 amp when the charger is placed back in FLOAT mode. By leaving the charger in Equalize mode until the charging current drops another 2 amps (i.e., from 5 amps down to 3 amps, as suggested by this question), there is no potential for damaging the battery due to "overcharging" it. Therefore, when the charger is placed back in FLOAT mode, there is no expectation that the FLOAT current will be higher than normal. Even if the Examinee were to believe that the small bit of overcharging could damage the battery, causing it to draw an excessive amount of current when the charger is placed back in FLOAT mode, this answer choice is nonetheless wrong because it suggests that the Tech Spec limit for this Div 1 Battery is "less than 1 amp"; it is not...the limit is 2 amps (refer to Tech Spec SR 4.8.2.1.a.1 for this value).

'C' is wrong: The two chargers for the bus may be unevenly loaded when the charger in placed back in FLOAT mode. This choice suggests that the charger that used for the EQUALIZE charge may somehow suffer a degraded performance when it's placed back in FLOAT mode; for example, if its output voltage could not be set at the ~ 132 VDC (nominal float voltage), then the other charger would have to carry more of the "float current load" on that bus.

DC Battery Fundamentals (such as discussed here) are generally not well-understood by Operators/Operator Candidates; for this reason, each of the distracters is sufficiently discriminating.

Question 9 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.00	
System ID:	1102907	
User-Defined ID:	REV 02, 01/06/15	
Cross Reference Number:	LGSOPS0095	
Topic:	Recognize potential consequence of overcharging a Div 1 Battery in Equalize mode	
Num Field 1:	2.6	
Num Field 2:	2.9	
Text Field:	263000 K5.01	
Comments:	Level	RO
	Tier	2
	Group	1
	KA # and Rating	263000 K5.01 (2.9/2.9)
	KA Statement	263000 DC Distribution Knowledge of the operational implications of the following concepts as they apply to D.C. ELECTRICAL DISTRIBUTION : K5.01 Hydrogen generation during battery charging
	References	DOE Electrical Science Handbook, Volume 2, June 1992 Revision
	Examinee References	None
	Learning Objective	LGSOPS0095, no specified objective
	Question source	New

	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.5
	Comments	Major revision to address discrimination validity concerns identified by Chief Examiner.

10 **ID: 1098329** **Points: 1.00**

Unit 1 plant conditions:

- OPCON 5*
- Core shuffle to final configuration is in progress per the Core Component Transfer Authorization sheet (CCTAS)

The following SRM count rate trends are observed between CCTAS steps #265 and #266:

Time	1A	1B	1C	1D
+1 min	70	42	20	55
+2 mins	120	100	45	120
+3 mins	162	102	102	165
+4 mins	200	103	160	175

Consider the following procedures:

- FH-105, Core Component Movement - Core Transfers
- ON-120, Fuel Handling Problems

WHICH ONE of the SRMs is indicating that an inadvertent criticality may be occurring, as defined by the above procedures?

A. 1A

B. 1B

C. 1C

D. 1D

Answer: C

Answer Explanation

Per FH-105 (Core Component Movement - Core Transfers), section 3.8.5.1, and ON-120 (Fuel Handling Problems), section 2.1, the doubling of an SRM count rate between two successive CCTAS steps is indication of an inadvertent criticality.

'C' is correct: 1C. Of the four SRMs, only the 1C count rate has doubled since the completion of CCTAS Step #266. At Time (completion of CCTAS Step #266) +1 minute, it reads 20 cpm; at Time +2 minutes, it reads 45 cpm (the first doubling); at Time +3 minutes, it reads 102 cpm (which is higher than 90 cpm which would be a second doubling).

'A', 'B', 'D' are wrong: 1A, 1B, 1D (respectively). Applying the same analysis to each of these other SRMs remaining revelas that none of them have experienced two doublings. Plausible to the examinee who either cannot recall the "two doublings" value for indication of an inadvertent criticality, or is less than competent in interpreting the SRM indications that result form the completion of CCTAS Step #266.

Question 10 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.50
System ID:	1098329
User-Defined ID:	REV 01, 12/15/14
Cross Reference Number:	LLOT0760.4C
Topic:	Recognize inadvertent criticality from SRM readings during refueling
Num Field 1:	3.4
Num Field 2:	3.9
Text Field:	234000 A1.03
Comments:	Level: RO Tier: 2 Group: 2 KA # and Rating: 234000 A1.03 (3.4/3.9) KA Statement: 234000 Fuel Handling Equipment Ability to predict and/or monitor changes in parameters associated with operating the FUEL HANDLING EQUIPMENT controls including: A1.03 core reactivity level References: FH-105, Rev.47; ON-120, Rev.24 Examinee References: None Learning Objective: LLOT0760.4C Question source: Bank 561061 Question history: None Cognitive level: Higher 10 CFR 55: 41.5 Comments: Minor revision to address NRC Rev.00 comments.

11 **ID: 1098330** **Points: 1.00**

Unit 1 is operating at 95% power, with the following:

- Both Recirc Pumps are running at 1295 rpm
- 1A ASD is operating with 3 cells bypassed on Phase 'A'

Then:

- One (1) more Phase 'A' cell fails on the 1A ASD

WHICH ONE of the following identifies:

(1) the response of the 1A Recirc Pump, and
(2) the required operator action?

A. (1) Trips
(2) Use Core Plate dP or Computer Point B017 to estimate Core Flow, then determine operating point on Power/Flow Map

B. (1) Trips
(2) Use Total Core Flow or Computer Point B018 to estimate Core Flow, then determine operating point on Power/Flow Map

C. (1) Speed lowers to between 1135 and 1195 rpm
(2) Verify Recirc Loop Flows are within 5% of each other

D. (1) Speed lowers to between 1135 and 1195 rpm
(2) Verify Recirc Loop Flows are within 10% of each other

Answer: B

Answer Explanation

Refer to S43.1.F, Attachment 2 FRR Alarm List, FRR # 2.1.1 (Cell Fault). If all 4 cells in a given phase are faulted/bypassed, an ASD trip occurs. Given these stem conditions, the result will be Single-Loop Operations (SLO), with 1B ASD (Recirc Pump) running at 1295 RPM. Operators enter OT-112, where Step 3.2 directs them to use Attachment 3 to determine the plant's operating point on the Power/Flow Map. Because the 1B Recirc Pump is operating at a speed >1000 rpm, the 2nd bullet of Step 3 applies...use the Core Flow recorder (XR-042-1R613) or computer point B018 to determine Core Flow.

'B' is correct: (1) Trips; (2) Use Total Core Flow or Computer Point B018 to estimate Core Flow, then determine operating point on Power/Flow Map. Correct for the reasons described above.

'A' is wrong: (1) Trips; (2) Use Core Plate dP or Computer Point B017 to estimate Core Flow, then determine operating point on Power/Flow Map. Plausible because it suggests the OT-112, Attachment 3, Step 3, 1st bullet action, which would be applicable if the 1B pump were operating at <1000 rpm.

'C' is wrong: (1) Speed lowers to between 1135 and 1195 rpm; (2) Verify Recirc Loop Flows are within 5% of each other. Plausible to the examinee who believes that the 1A ASD is still running but at a lower speed (which would be the case if something less than all 4 cells had failed). In that case, the examinee would consider only the need to check the resulting Recirc Loop Flow mismatch that results (i.e., the approximate 100-150 rpm drop on the 1A ASD due to the cell failure, leaving the 1B ASD operating at the initial 1295 rpm) to verify that the applicable mismatch limit of Tech Spec LCO 3.4.1.3 is still satisfied. Per that LCO, the mismatch limit is 5% of rated core flow when the Total Core Flow is at or above 70% of rated core flow. This 5% limit would apply in the case of this question, being that the resulting Total Core Flow (after the 1A ASD speed drop) is still well above 70% of rated.

'D' is wrong: (1) Speed lowers to between 1135 and 1195 rpm; (2) Verify Recirc Loop Flows are within 10% of each other. Plausible for reasons almost identical of that for choice 'C'. In this case, however, the examinee either believes that the resulting Total Core Flow (after the 1A ASD speed drop) is below 70% of rated (in which case the 10% mismatch limit of LCO 3.4.1.3 applies), or simply confuses the two mismatch limits.

Question 11 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	5	
Difficulty:	3.00	
System ID:	1098330	
User-Defined ID:	REV 02, 01/06/15	
Cross Reference Number:	LGSOPS1540.2	
Topic:	Determine OT-112 actions based on resulting core flow after ASD trip	
Num Field 1:	3.1	
Num Field 2:	3.2	
Text Field:	295001 AA1.07	
Comments:	Level	RO
	Tier	1
	Group	1
	KA # and Rating	295001 AA1.07 (3.1/3.2)
	KA Statement	295001 Partial or Complete Loss of Forced Core Flow Circulation Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: AA1.07 Nuclear boiler instrumentation system
	References	OT-112, Rev.52 S43.1.F, Rev.1 U/1 Tech Spec 3.4.1.3 (latest)
	Examinee References	None
	Learning Objective	LGSOPS1540.2
	Question source	Bank 974482
	Question history	None
	Cognitive level	Higher

	10 CFR 55 Comments	41.7, 41.10 Minor editorial to match choice 'A' and 'B' Part (2) symmetries. Minor revision again (Rev.02) to two distractors, to address NRC Rev.01 comments.

12 **ID: 1098331** **Points: 1.00**

Unit 1 is operating at 100% power with a normal electrical lineup.

A ground fault occurs on the 11 Unit Aux Bus.

WHICH ONE of the following identifies the automatic plant response, assuming no operator action?

A. Reactor scrams due to low RPV water level

B. Reactor scrams due to a main turbine trip

C. Reactor remains operating but at a significantly lower power

D. Reactor remains operating but at a lower main condenser vacuum

Answer: A

Answer Explanation

Refer to drawing E-0001, Sheet 1 (Single-Line Diagram - Station). With the Main Generator on the grid, 11 Unit Aux Bus is normally powered from the 11 Unit Aux Transformer via breaker 252-10113. A ground fault on the 11 Unit Aux Bus trips and locks out the 252-10113 breaker and prevents closure of the alternate supply breakers 252-10102 (from the 10 Startup Bus) and 252-10106 (from the 20 Startup Bus), de-energizing all 11 Unit Aux Bus loads. The following important loads trip on the loss of the 11 Unit Aux Bus (refer to alarm response card ARC-MCR-126, B1 to validate): 1A ASD/Recirc Pump; 1A and 1C Circ Water Pumps; and 1A and 1C Condensate Pumps. While the the 1A Recirc Pump trip will result in Single-Loop Operations (SLO), it will not produce an automatic reactor scram. While the trip of two running Circ Water Pumps (leaving only two CW Pumps running) might otherwise cause main condenser vacuum to degrade, the Recirc Pump trip causes a more-than-ample reduction in reactor power (to about 55%) that will preclude the drop in main condenser vacuum. The limiting concern, then, is the trip of the two Condensate Pumps (1A and 1C). When they trip, the Reactor Feed Pumps (RPFs) trip (in a time-delayed sequence) on low suction pressure, resulting in a rapid drop in RPV water level and an automatic scram at +12.5" (low RPV water level).

'A' is correct: Reactor scrams due to low RPV water level. Correct for the reasons described above.

'B' is wrong: Reactor scrams due to a main turbine trip. Plausible to the examinee who confuses the 11 Unit Aux Bus, which powers two of the condensate Pumps, with the 12 Unit Aux Bus, which powers only one Condensate Pump. In that case, the examinee believes that only the 1B Condensate Pump has tripped. If that were the case, an automatic Recirc Pump runback would occur, but so too would the trip of two Circ Water Pumps, causing main condenser vacuum to drop to the point of a main turbne trip on low vacuum (at 21.5" Hg Vac).

'C' is wrong: Reactor remains operating but at a significantly lower power. Plausible to the examine who is able to determine all of the major pumps that trip (as identified above), but who incorrectly concludes that the Recirc Pump trip produces a large enough power reduction so as to preclude the loss of any more than the 1C RFP on a low suction pressure trip. That examinee recalls that the low suction pressure trip for the RFPs is time-delay staggered...1C trip first, followed by the 1B, 5 seconds later, then the 1A, 5 seconds after that...the examinee incorrectly concludes that the unloading of the 1C RFP from the single, running Condensate Pump (1B), allows the 1A and 1B RFP to remain in operating, thereby preventing RPV water level from lowering to the +12.5" low level scram setpoint.

'D' is wrong: Reactor remains operating but at a lower main condenser vacuum. Plausible for reasons similar to that of choice 'A'. In this case, the examinee believes that only one RFP will trip on low suction pressure, and forgets that the 1A ASD/Recirc Pump tripped. This examinee concludes that reactor power remains essentially unchanged, but with the plant operating at a degraded vacuum due to the dual-Circ Water Pump trip. Since the stem asks for the "automatic plant response", this examinee determines that the reactor will in fact remain operating, allowing ample time for operators to enter OT-116 (Loss of Condenser Vacuum) and mitigate the degrading vacuum by manually reducing power using Recirc.

Question 12 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	3.00	
System ID:	1098331	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LLOT0540.13E	
Topic:	Loss of 11 Unit Aux Bus - Predict plant response and the reason for that response	
Num Field 1:	3.7	
Num Field 2:	3.7	
Text Field:	295003 AK3.05	
Comments:	Level	RO
	Tier	1
	Group	1
	KA # and Rating	295003 K3.05 (3.7/3.7)*
	KA Statement	295003 Partial or Complete Loss of AC Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : AK3.05 Reactor SCRAM
	References	E-0001, Sheet 1, Rev.29 ARC-MCR-126, B1, Rev.2 OT-116, Rev.37
	Examinee References	None
	Learning Objective	LLOT0540.13e
	Question source	New
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.5

	Comments	*Although the selected KA is a "reason for the [Scram] response", we've framed the question in such a way that only the correct answer ('A') and a similarly constructed distracter ('B') suggest that "reason." To optimize the psychometrics, we've intentionally omitted the "reason" aspect from the remaining distracters ('C' and 'D'), as well as from the question statement, itself. Because the correct answer choice does address the "reason" aspect (and is balanced by choice 'B'), this construct adequately provides the required KA match. --- _ Minor editorial change to stem to address NRC Rev.00 comments.

13 **ID: 1098336** **Points: 1.00**

Unit 2 refueling outage is in progress.

Unit 1 plant startup is in progress when a fire causes operators to evacuate the MCR.

Operators commence a plant cooldown from the RSP, with the following:

- RPV pressure indicates 685 psig
- RPV water level indicates +30"

(Assume indicated RPV water level is held constant and cooldown rate is established at **maximium permitted by SE-1, Remote Shutdown**.)

WHICH ONE of the following identifies the **EARLIEST** approximate time from now that Shutdown Cooling can be initiated, and how Wide Range RPV water level indication will then compare to actual level?

	SDC Initiation	Wide Range Level
A.	2.3 hours from now	Lower than actual
B.	1.8 hours from now	Lower than actual
C.	1.8 hours from now	Higher than actual
D.	2.3 hours from now	Higher than actual

Answer: C

Answer Explanation

Per the Steam Tables: at 685 psig (700 psia), saturation temperature is 503°F.
The Shutdown Cooling (SDC) RPV pressure interlock is 75 psig (90 psia) (see SE-1, step 4.9.1); its saturation temperature is 320°F. The maximum permitted cooldown rate is 100°F/hr (see SE-1, step 4.9.13). Cooling down at 100°F/hr, it will take approximately 1.83 hours (~1.8) before SDC can be initiated.

The Remote Shutdown Panel (RSP) uses Wide Range level indication, which is calibrated for "hot" conditions (rated pressure). Sensor/Detector Fundamentals reminds us that "hot calibrated indications read HIGHER than actual when operated at cold conditions." Thus, RSP level will be HIGHER than actual when operators place SDC in service.

'C' is correct: 1.8 hours from now; Higher than actual. Correct for the above reasons.

'A' is wrong: 2.3 hours from now; Lower than actual. The first part is plausible to the examinee who incorrectly recalls the cooldown rate permitted by SE-1; instead, that examinee believes that the 80°F/hr cooldown rate applies (this is the maximum permitted cooldown rate (an administrative limit) found in GP-3, Appendix 1 ("Establishing Cold Shutdown"), Section 3.1.64. In this case, the examinee divides the 183°F total cooldown (from 503°F down to 320°F) by 80 and gets a result of 2.28 hours (~2.3). The second part is plausible to the examinee who cannot recall the Generic Fundamentals regarding the inaccuracy of RPV level instruments when operating at other than their calibrated conditions.

'B' is wrong: 1.8 hours from now; Lower than actual. The second part is plausible for the same reason as for chocie 'A'.

'D' is wrong: 2.3 hours from now; Higher than actual. The first part is plausible for the same reason as for choice 'A'.

Question 13 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	5	
Difficulty:	2.50	
System ID:	1098336	
User-Defined ID:	REV 02, 01/06/15	
Cross Reference Number:	LGSOPS2000.04	
Topic:	SE-1 - Plant Cooldown, Determine Time Until SDC is permitted	
Num Field 1:	3.3	
Num Field 2:	3.5	
Text Field:	295016 AA2.06	
Comments:	Level	RO
	Tier	1
	Group	1
	KA # and Rating	295016 AA2.06 (3.3/3.5)
	KA Statement	295016 Control Room Abandonment Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: AA2.06 Cooldown rate
	References	SE-1, Rev.70
	Examinee References	Steam Tables (per Chief Examiner: is not counted as an "open reference")
	Learning Objective	LGSOPS2000.04
	Question source	New*
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.10
	Comments	*Major revision to address NRC Rev.00 comments. Was a "Bank" item, is now accounted for as a "New" item. Editorial change to answer choices, to address NRC Rev.01 comments.

14 **ID: 1098344** **Points: 1.00**

Unit 2 is operating at 100% power.

WHICH ONE of the following, by itself, requires a Unit 2 Tech Spec LCO Action that is 1-hour or less?

A. A failed interlock allows both airlock doors to be opened at the same time (R.E. 217' NW)

B. The 2B Recirc Pump motor breaker trips on an electrical fault

C. The Div 3 125 VDC Battery Charger is declared inoperable

D. An EHC malfunction causes RPV pressure to stabilize at 1060 psig

Answer: D

Answer Explanation

Refer to the following Unit 2 (U/2) Tech Specs:

- TS 3.4.1.1.a (one recirc loop not in service) is a "within 4 hours" action
- TS 3.4.6.2 (reactor pressure above 1053 psig) is a "within 15 minutes" action
- TS 3.6.5.1.1 (reactor enclosure secondary containment integrity not maintained) is a "within 4 hours" action
- TS 3.8.2.1.a.1 (one charger inoperable) is a "within 2 hours" action

'D' is correct: An EHC malfunction causes RPV pressure to stabilize at 1060 psig. Correct per TS 3.4.6.2 Action, as described above.

'A' is wrong: A failed interlock allows both airlock doors to be opened at the same time (R.E. 217' NW).

'B' is wrong: The 2B Recirc Pump motor breaker trips on an electrical fault.

'C' is wrong: The Div 3 125 VDC Battery Charger is declared inoperable.

All three distracters suggest fairly short Action Times (4 hours or less); therefore, each is plausible when asked "from-memory".

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 14 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.50
System ID:	1098344
User-Defined ID:	REV 01, 12/15/14
Cross Reference Number:	LLOT0041B.9
Topic:	High RPV Pressure - Recognize condition requiring 1-hour TS Action
Num Field 1:	3.9
Num Field 2:	4.5
Text Field:	295025 2.2.39
Comments:	Level: RO Tier: 1 Group: 1 KA # and Rating: 295025 2.2.39 (3.9/4.5) KA Statement: 295025 High Reactor Pressure 2.2.39 Knowledge of less than or equal to one hour Technical Specification action statements for systems References: U/2 Tech Spec 3.4.6.2 (latest) U/2 Tech Spec 3.1.3.6 (latest) U/2 Tech Spec 3.4.1.1 (latest) U/2 Tech Spec 3.8.2.1 (latest) Examinee References: None Learning Objective: LLOT0041B.9 Question source: New Question history: None Cognitive level: Lower 10 CFR 55: 41.7 Comments: Minor editorial changes to answer choices to address NRC Rev.00 comments.

15 **ID: 1098353** **Points: 1.00**

WHICH ONE of the following identifies a possible automatic actuation of a Unit 1 fire protection (fire suppression) system?

A. Heat detectors sense a fire in the Main Generator Exciter

B. Heat detectors sense a fire in the Main Turbine Lube Oil Reservoir

C. Smoke detectors sense a fire in the RCIC Pump Room

D. Smoke detectors sense a fire in the Remote Shutdown Panel Room

Answer: B

Answer Explanation

With only one exception, heat detectors are used to automatically actuate fire protection systems (sprinklers and sprays), but smoke detectors are used solely to actuate Fire Alarm Panel alarms. The one exception is: Control Building Elevation 217' Switchgear I and II, which uses smoke detectors to automatically actuate Pre-Action Sprinklers (refer to alarm response card ARC-MCR-006, J-2-U, to validate this).

'B' is correct: Heat detectors sense a fire in the Main Turbine Lube Oil Reservoir. Per ARC-MCR-006, B-4-U, heat detector devices 1HD936-1 thru 936-8 are used to automatically actuate wet pipe and deluge sprinkler systems.

'A' is wrong: Heat detectors sense a fire in the Main Generator Exciter. Per ARC-MCR-006, H-5-L, the Exciter does use heat detectors (as well as smoke detectors), but for alarm only. There is NO automatic actuation of any type of fire protection (suppression) system. Plausible to the examinee who does recall that, with the one exception, heat detectors are used for actuating these systems; however, the examinee fails to recall that the Exciter does not have any automatic actuation system.

'C' is wrong: Smoke detectors sense a fire in the RCIC Pump Room. Per ARC-MCR-006, L-1-L, the RCIC Pump Room does have smoke detectors, but they are for alarm only; there are NO automatic fire suppression systems of any type. Plausible to the examinee who confuses smoke detectors with heat detectors, mistakenly believing that smoke detectors rather than heat detectors are the usual type of detector used for actuating fire suppression systems.

'D' is wrong: Smoke detectors sense a fire in the Remote Shutdown Panel Room. Per ARC-MCR-006, J-3-L, the Remote Shutdown Panel (RSP) Room uses both smoke detectors and heat detectors. While the heat detectors do automatically initiate the HALON fire suppression system, the smoke detectors are used for alarm only. Plausible to the examinee who vaguely recalls that the RSP room does have an automatic Halon system but who cannot recall which detector type is used for the automatic actuation.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 15 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	3.00	
System ID:	1098353	
User-Defined ID:	REV 02, 01/06/15	
Cross Reference Number:	LGSOPS0022.5	
Topic:	Recall relationship between fire detectors and fire water systems	
Num Field 1:	2.6	
Num Field 2:	2.7	
Text Field:	600000 AK2.01	
Comments:	Level	RO
	Tier	1
	Group	1
	KA # and Rating	600000 AK2.01 (2.6/2.7)
	KA Statement	600000 Plant Fire On Site Knowledge of the interrelations between PLANT FIRE ON SITE and the following: AK2.01 Sensors / detectors and valves
	References	ARC-MCR-006, B-4-U, Rev.1 ARC-MCR-006, H-5-L, Rev.3 ARC-MCR-006, L-1-L, Rev.1 ARC-MCR-006, J-3-L, Rev.6 ARC-MCR-006, J-2-U, Rev.1
	Examinee References	None
	Learning Objective	LGSOPS0022.5
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.4
	Comments	Major revision to stem and answer (Rev.02) choices to address NRC Rev.00 comments.

16 **ID: 1098424** **Points: 1.00**

Plant conditions during the Summer with heavy grid loading conditions:

- Unit 1 is in OPCON 4 (forced outage)
- Unit 2 is operating at 100% power
- Normal electrical lineup for these Unit conditions

TSO informs LGS of the following:

- The Post Trip Contingency Voltage Drop (PTCVD) for the 230 KV system has exceeded its limit

Operators enter E-5 (Grid Emergency) and, as directed by that procedure, perform the following:

- Take the actions to "harden" the 10 Bus

WHICH ONE of the following describes:

(1) the action of "hardening" the 10 Bus, and
(2) the reason for taking that action?

A. (1) Removes certain 13.2 KV loads from the 12 Unit Aux Bus
(2) Help ensure that the grid can handle the loads on the Unit 2 Safeguard Buses should that Unit scram

B. (1) Removes certain 13.2 KV loads from the 11 Unit Aux Bus
(2) Minimize the risk of losing Shutdown Cooling on Unit 1

C. (1) Removes certain 13.2 KV loads from the 11 Unit Aux Bus
(2) Help ensure that the grid can handle the loads on the Unit 2 Safeguard Buses should that Unit scram

D. (1) Removes certain 13.2 KV loads from the 12 Unit Aux Bus
(2) Minimize the risk of losing Shutdown Cooling on Unit 1

Answer: C

Answer Explanation

Refer to the E-5 (Grid Emergency) Bases; with respect to this question, these are the key sections/steps:

- 3.13.2 addresses exceeding the PTCVD limits for the 10 Bus; specifically, 3.13.2.3 directs operators to take the actions of Attachment 1
- Attachment 1, step 2.1.4, directs operators to consider the realigning safety buses and/or performing the hardening actions per Attachment 3 (in the case of this question...for the 10 Bus)
- The "bases" statement directly below Attachment 1, step 2.1.4, describes the reason for 2.1.4 as: "improving the margin for the PTCVD percentage"
- Attachment 3 is all about "Actions to Harden the 10 Bus to improve PTCVD limits" with one unit shutdown and one unit operating
- Regardless of the current safety bus alignment for the Units (not defined for the stem of this question), the Attachment 3 actions, in either case, involve removing one or more 13.2 KV loads (Condensate Pump and/or Circ Water Pump) from the 10 Bus

- The "Bases" discussion at the bottom of Attachment 1, page 1, describes the operational concern regarding "inadequate contingency voltage" (which is the concern when it's been determined that either the 230 KV or 500 KV system is exceeding its PTCVD limits)
- The "Bases" discussion below Table 1 of step 3.13.2 (in the procedure body) clarifies this concern a bit more..."If LGS post trip percentage voltage drop is in excess of the values used in the Voltage Regulation study then the analysis for safeguard bus loading cannot be assured"
- The "Bases" discussion below step 1.4 (in the procedure body) adds more clarification: "The Limerick Voltage regulation study includes various assumptions on the impacts to the transmission system following the trip of an LGS unit..."

Given all of the above information, here is the single concept to be understood (explained in a way that addresses this question's stem conditions and answer choices):

We "harden" the 10 Bus by removing unneeded 13.2 KV loads belonging to the shutdown Unit (Unit 1) that are being fed from the 10 Bus. The "normal electrical lineup" (in the stem) means that **the 10 Bus is powering the 11 Unit Aux (13.2 KV) Bus.** It also means that the 12 Unit Aux Bus is being powered from the 20 Bus (which is NOT a concern with respect to this question). [Refer to 1S91.9.A (COL), page 3, steps 5, 6, 11, and 12, to validate this.]

The reason for hardening the 10 Bus is to "lighten" the load on that bus, and in-turn lighten the load on the 230 KV system, so as to allow that system to operate at a somewhat higher voltage (i.e., increasing the voltage drop margin). This acknowledges that, should Unit 2 trip (scram), it may result in a decrease in 230 KV system voltage (on an already heavily-loaded grid system). **Having the improved PTCVD margin beforehand will help ensure that the offsite power system will still be able to safely handle the loading of the Unit 2 safeguards (safe shutdown) buses.**

'C' is correct: (1) Removes certain 13.2 KV loads from the 11 Unit Aux Bus; (2) Help ensure that the grid can handle the loads on the Unit 2 Safeguard Buses should that Unit scram. Correct for the reasons described above.

'A' is wrong: (1) Removes certain 13.2 KV loads from the 12 Unit Aux Bus; (2) Help ensure that the grid can handle the loads on the Unit 2 Safeguard Buses should that Unit scram. Part (1) is wrong, but plausible to the examinee who cannot recall the normal 10 Bus / 20 Bus alignment already discussed. That examinee incorrectly determines that the 12 Unit Aux Bus is being fed from the 10 Bus; it is not.

'B' is wrong: (1) Removes certain 13.2 KV loads from the 11 Unit Aux Bus; (2) Minimize the risk of losing Shutdown Cooling on Unit 1. Part (2) is wrong, but plausible to the examinee who is distracted by the fact that maintaining Shutdown Cooling in operation is always extremely important (as it is here for Unit 1). While such is true, it is not the reason for any of the actions directed by the E-5 procedure.

'D' is wrong: (1) Removes certain 13.2 KV loads from the 12 Unit Aux Bus; (2) Minimize the risk of losing Shutdown Cooling on Unit 1. Both Parts are wrong, but plausible for the same reasons as for choices 'A' and 'B'.

Question 16 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.50
System ID:	1098424
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LLOT1566.02
Topic:	Recall action and reason for that action of hardening 10 Bus per E-5, Grid Emergency
Num Field 1:	3.6
Num Field 2:	3.9
Text Field:	700000 AK3.02
Comments:	Level: RO Tier: 1 Group: 1 KA # and Rating: 700000 AK3.02 (3.6/3.9) KA Statement: 700000 Generator Voltage and Electric Grid Disturbances Knowledge of the reasons for the following responses | as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID | DISTURBANCES: AK3.02 Actions contained in abnormal operating procedure for voltage and grid disturbances References: E-5, Rev.21 E-5 Bases, Rev.21 E-0001, Sheet 1, Rev.29 1S91.9.A (COL), Rev.1 Examinee References: None Learning Objective: LLOT1566.02 (in lesson plan LGSOPS2000) Question source: New Question history: None Cognitive level: Higher* 10 CFR 55: 41.4, 41.10 Comments: * Justification for Higher Cognitive categorization: Part (1) of each answer choice requires the examinee to recognize the 10 Bus / 20 Bus alignments that must exist with Unit 1 shutdown and Unit 2 operating.

17 **ID: 1098425** **Points: 1.00**

Unit 1 has been operating at 100% power (steady) for 30 days, with the following:

- Reactor pressure is reading 1044.5 psig on DEHC HMI display

Reactor pressure rises to 1050 psig (unexplained rise on DEHC HMI) and stabilizes there.

WHICH ONE of the following DEHC HMI display indications could be associated with this event?

A. Throttle Pressure reading 985 psig

B. Pressure Set reading 955 psig

C. All CV positions reading 46%

D. All BPV positions reading 10%

Answer: C

Answer Explanation

Implication of the stem conditions indicating that the Unit has been operating at 100%, steady-state for 30 days: that operators have long-ago completed the "Power Ascension to Rated Power" actions of GP-2, Section 3.5.67 (refer to that procedure section). Initially (before the unexplained pressure rise), then, a reactor pressure reading of 1044.5 psig is not only "normal" but also within the band specified by Step 3.5.67.1. Per 3.5.67.2, Unit 1 average turbine control valve (CV) positions must be >/= 50% open. Per 3.5.64, Unit 1 Pressure Set must be between 955 psig and 970 psig with the plant operating at 100% power; it is actually set at 955 psig. Examinee is expected to recognize that a "normal" Throttle Pressure (i.e., PAM pressure) with a 955 psig Pressure Set when operating at rated power conditions is approximately 985 psig (assuming an approximate reactor pressure of 1044-1045 psig).

'C' is correct: All CV positions reading 46%. The Unit 1 CV's are not throttled open as far they should be (>/= 50%, as already discussed). For whatever reason they may have throttled partially closed, the result will be a rise in reactor pressure.

'A' is wrong: Throttle Pressure reading 985 psig. As described above, this is a "normal" throttle pressure given the stem conditions. Plausible to the examinee who fails to recognize this fact.

'B' is wrong: Pressure Set reading 955 psig. Not only is a Pressure Set of 955 psig within the required band of GP-2, step 3.5.64, it is in fact the normal setting. As such, the CV positions would be reading something greater than their normal positions (i.e., something >50% open), resulting in reactor pressure drop, not rise. Plausible to the examinee who doesn't comprehend the relationship between Pressure Set and CV response as a result of a change in Pressure Set.

'D' is wrong: All BPV positions reading 10%. If the system/plant were behaving normally (i.e., plant conditions before the pressure rise event), all 9 turbine bypass valves (BPVs) would be fully closed (reading 0% open). Per GP-2, step 3.5.67.8, the Unit 1 CVs should never be more than about 58% open (which is well short of their 100% open position). What this means is that, assuming that the CVs themselves are not the direct cause of the reactor pressure rise, a 5 psig rise in reactor pressure would require the CVs to throttle open just a few percent more (if that). There would not be the need for the Bypass Control Unit to develop a signal to open the BPVs. Plausible to the examinee who has a weak comprehension of the relationship between the Pressure Control Unit and the Bypass Control Unit of EHC.

Question 17 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	2.50
System ID:	1098425
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	NONE
Topic:	Recognize MCR indication associated with unexplained rise in RPV pressure
Num Field 1:	4.6
Num Field 2:	4.3
Text Field:	295007 2.1.31
Comments:	Level: RO Tier: 1 Group: 2 KA # and Rating: 295007 2.1.31 (4.6/4.3) KA Statement: 295007 High Reactor Pressure 2.1.31 Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup. References: GP-2, Rev.153 Examinee References: None Learning Objective: No specified objective Question source: New Question history: None Cognitive level: Higher 10 CFR 55: 41.5, 41.7, 41.10 Comments: None

18 **ID: 1098446** **Points: 1.00**

Unit 2 is operating at 100% power when the following occurs:

- Drywell pressure is indicating 0.5 psig, up slow (~ 0.1 psig every 5 minutes)

Operators enter OT-101 (High Drywell Pressure) and observe the following in the MCR:

- 2A Recirc Pump seal pressures read 1035 psig / 520 psig (#1 / #2, respectively)
- 2B Recirc Pump seal pressures read 1030 psig / 510 psig (#1 / #2, respectively)
- 2A DWCW Pump - Green light ON, Red light OFF
- 2B DWCW Pump - Green light OFF, Red light ON
- SV-59-128A, Inst Gas 'A' - Green light ON, Red light OFF
- SV-59-128B, Inst Gas 'B' - Green light ON, Red light OFF
- N2 Purge Flow reads 0 SCFM
- N2 Makeup Flow reads 0 SCFM

WHICH ONE of the following OT-101 mitigation strategies is required?

A. Attachment 1 - Recirc Pump Seals

B. Attachment 2 - Instrument Air Backing Up Instrument Gas

C. Attachment 3 - Loss of Drywell Cooling

D. Attachment 4 - Unidentified or RWCU Cause of Rising Drywell Pressure

Answer: C

Answer Explanation

Per OT-101, step 3.2, operators will diagnose the source of the drywell pressure trend. For the sake of these stem conditions, the following key plant parameters are considered in step 3.2: Recirc Pump seal pressures; and Drywell Chill Water (DWCW) system operation. Once the source of the problem has been determined, step 3.3 directs operators to execute the appropriate Attachment. There are 4 such Attachments: Attachment 1 mitigates the problem of failed Recirc Pump seals; Attachment 2 mitigates the problem of Instrument Air being introduced into the otherwise inerted drywell via the Instrument Gas headers; Attachment 3 mitigates the problem of a loss of drywell cooling; Attachment 4 mitigates the problem of either RWCU leakage into the drywell, or of having determined that the source is unknown.

Consider the MCR indications given in the stem:

- The seal pressures for both Recirc Pumps are reading "normal" for rated power conditions (i.e., #1 pressure is approximately RPV pressure, #2 pressure is approximately one-half of RPV pressure)

- Both normally-closed block valves that connect Instrument Air to the Instrument Gas system are indicating CLOSED (i.e., Green light ON, Red light OFF)

- Nitrogen flows (both Purge and Makeup) to the Primary Containment are both indicating 0 scfm (i.e., no valve leakage and/or flow controller malfunction is causing an unintentional de-inerting of the containment)

- However, the 2A DWCW (Circ) Pump is not running (Green light ON, Red light OFF); this is

indicative of a partial loss of drywell cooling (i.e., normally, both pumps are running in order to maintain a "normal", rated power, drywell temperature (and therefore drywell pressure))

'C' is correct: Attachment 3 - Loss of Drywell Cooling. This is the mitigation strategy that satisfiies the direction given by OT-101, step 3.3.

'A' is wrong: Attachment 1 - Recirc Pump Seals. Plausible to the examinee who cannot effectively diagnose seal failures given the seal pressures.

'B' is wrong: Attachment 2 - Instrument Air Backing Up Instrument Gas. As discussed above, there is no indication (in the stem) that Instrument Air is leaking into the drywell. Plausible to the examinee who doesn't comprehend the significance of the given N2 flows (i.e., that these same recorder points, normally used for monitoring nitrogen flow into the primary containment, are also used to monitor Instrument Air flow into the containment when required, such as when de-inerting the containment).

'D' is wrong: Attachment 4 - Unidentified or RWCU Cause of Rising Drywell Pressure. Because operators have in fact determined the source of the drywell pressure trend (i.e., some loss of drywell cooling), employing Attachment 4 is not required. Additionally, there is no stem information regarding the status of RWCU; therefore, neither would they be required to execute Attachment 4 to mitigate an RWCU leak into the drywell.

Question 18 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	2.50	
System ID:	1098446	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	LGSOPS1540.3	
Topic:	Determine required OT-101 strategy for an upward trend of drywell pressure	
Num Field 1:	3.2	
Num Field 2:	3.4	
Text Field:	295010 AK1.03	
Comments:	Level	RO
	Tier	1
	Group	2
	KA # and Rating	295010 AK1.03 (3.2/3.4)
	KA Statement	295010 High Drywell Pressure Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE: AK1.03 Temperature increases
	References	OT-101, Rev.35
	Examinee References	None
	Learning Objective	LGSOPS1540.3
	Question source	New
	Question history	None

	Cognitive level	Higher
	10 CFR 55	41.10
	Comments	None

19 **ID: 1098470** **Points: 1.00**

Unit 1 plant conditions:

- RPV water level is -150", steady
- Core Spray is maintaining RPV water level
- LPCI 'A' Injection Valve is overridden CLOSED (P-T-L)
- RHR 'B' is operating in Drywell Spray
- Suppression Pool level is 30 feet

PRO is directed to lower Suppression Pool level using T-233 (RHR to Radwaste).

WHICH ONE of the following describes an action that the PRO performs in the MCR to complete this task?

A. OPEN the RHR 'A' HX Inlet (Outlet) Valves, F047A (F003A)

B. START 'A' or 'C' RHRSW Pump

C. Re-align Drywell Sprays to RHR 'A'

D. START 'B' or 'D' RHRSW Pump

Answer: ~~B~~ KO 3/10/15 Accept A or B

Answer Explanation

Refer to T-233 (its use being directed from T-102 (Primary Containment Control, Step SP/L-11)...Only RHR 'A' is capable of being aligned to Radwaste in order to reduce Suppression Pool water level. Reviewing the actions of Section 4.0, we discover the following (with respect only to the answer choices for this question):

Step 4.1 is a MCR action but is not required because the RHR 'A' HX Inlet (Outlet) Valves are already open (normally-open valves), and nothing in the stem conditions indicates any reason why operators would have been directed to close them before the CRS directed this evolution.

Step 4.4 is a MCR action and will have to be performed. The RHRSW Pumps have no auto-start features; therefore, they did not auto-start in response to the -129" LOCA signal. MCR operators will have to place either the 'A' or 'C' RHRSW Pump in service.

Step 4.5 is a MCR action but is not required because the 'A' RHR Pump is already running...it auto-started on the -129" LOCA signal. The fact that operators have overridden CLOSED (Pull-To-Lock) its associated LPCI Injection Valve (F017A) indicates that the RHR 'A' Pump is still running (on min flow).

'B' is correct: START 'A' or 'C' RHRSW Pump. Correct for the reasons described above.

'A' is wrong: OPEN the RHR 'A' HX Inlet (Outlet) Valves, F047A (F003A). Plausible to the ~~examiner who~~ forgets that these are normally-open valves ~~or who believes~~ that one or both have auto-closed in response ~~to the LOCA~~ signal. KO 3/10/15 Accept A or B

'C' is wrong: Re-align Drywell Sprays to RHR 'A'. Plausible to the examinee who cannot recall whether T-233 uses RHR 'A' exclusively (it does) or rather RHR 'B' exclusively (it doesn't). That examinee believes it to be RHR 'B'. Recognizing that RHR 'B' is currently spraying the drywell, the operator would have to swap drywell sprays over to RHR 'A'.

'D' is wrong: START 'B' or 'D' RHRSW Pump. Plausible for reasons similar to those of choice 'C'. In this case, however, the examinee also forgets that one of the 'B' Loop RHRSW pumps was already started when RHR 'B' was placed in drywell sprays.

Question 19 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	1098470
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LLOT0051.IL10
Topic:	Determine actions to lower Supp Pool level with LOCA signal present using T-233
Num Field 1:	2.9
Num Field 2:	3.0
Text Field:	295029 EA1.03
Comments:	Level RO Tier 1 Group 2 KA # and Rating 295029 EA1.03 (2.9/3.0) KA Statement 295029 High Suppression Pool Water Level Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: EA1.03 RHR/LPCI References T-102, Rev.24 T-225 U/1, Rev.22 T-233 U/1, Rev.14 Examinee References None Learning Objective LLOT0051.IL10 Question source New Question history None Cognitive level Higher 10 CFR 55 41.7, 41.10 Comments None

20 **ID: 1098495** **Points: 1.00**

Unit 1 plant conditions:

- HPCI Pump Room temperature is 170°F
- RCIC Pump Room temperature is 150°F
- 1A/1C RHR Pump Room temperature is 160°F
- 1B/1D RHR Pump Room temperature is 150°F

WHICH ONE of the following identifies **ALL** of the systems/subsystems (from the above list) that are available by procedure for safe shutdown of the plant?

A. RCIC

B. RCIC
1A/1C RHR

C. 1A/1C RHR
1B/1D RHR

D. HPCI
RCIC

Answer: D

Answer Explanation

Refer to T-103 (Secondary Containment Control), Table SCC-2 (Max Safe Operating (MSO) Temperature Values). HPCI MSO is 176°F; RCIC MSO is 155°F; 1A/1C RHR and 1B/1D RHR MSO's are 140°F. The stem conditions indicate that the following systems/subsystems are BELOW their MSO temperatures: HPCI and RCIC. Per the T-103 Bases, page 4 of 32..."The MSO, as used in T-103, is defined as the highest value of a parameter at which neither (1) equipment necessary for the safe shutdown of the plant will fail, nor (2) personnel access necessary for the safe shutdown of the plant will be precluded."

'D' is correct: HPCI, RCIC. Among the 4 given systems/subsystems, only HPCI and RCIC are below MSO and are, therefore, available for safe shutdown of the plant.

'A', 'B', 'C' are wrong. Each of these choices suggest one or more systems/subsystems for which the temperature is ABOVE the MSO; therefore, the system(s) is/are NOT available for safe shutdown of the plant.

NOTE - Validation proved that recognizing the HPCI/RCIC room MSO values (from memory) is reasonable for an examinee, even more so for a well-prepared, recently-trained ILT RO Candidate.

Question 20 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	3.00	
System ID:	1098495	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LLOT1560.5	
Topic:	Given T-103 temperatures, identify systems that are available for safe shutdown of the plant	
Num Field 1:	3.3	
Num Field 2:	3.5	
Text Field:	295032 EA2.02	
Comments:	Level	RO
	Tier	1
	Group	2
	KA # and Rating	295032 EA2.02 (3.3/3.5)
	KA Statement	295032 High Secondary Containment Area Temperature Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: EA2.02 Equipment operability
	References	T-103, Rev.20
	Examinee References	None
	Learning Objective	LLOT1560.5
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.7, 41.10
	Comments	Minor editorial to explanation to address NRC Rev.00 comments.

21 **ID: 1098544** **Points: 1.00**

Unit 1 plant startup is in progress.

All IRMs are on Range 3 with the following indications:

A - 80/125
B - 72/125
C - 78/125
D - 124/125
E - 75/125
F - 83/125
G - 122/125
H - 74/125

WHICH ONE of the following describes the plant response related to the rod withdraw block and scram functions?

A. Rod withdraw block, only

B. Rod withdraw block and RPS 'A' half-scram, only

C. Rod withdraw block and RPS 'B' half-scram, only

D. Rod withdraw block and full scram

Answer: D

Answer Explanation

Use alarm response card ARC-MCR-107, F3 to validate the following: IRM Upscale rod block setpoint is 85/125 of scale on any one of the 8 IRM channels. Use ARC-MCR-107, H3 to validate the following: IRM Upscale trip (RPS actuation) setpoint is 120/125 of scale. IRMs A, C, E, and G input to RPS Trip System 'A'; IRMs B, D, F, and H input to RPS Trip System 'B'. IRM trips are enabled so long as the Reactor Mode Switch is NOT in RUN.

'D' is correct: Rod withdraw block and full scram. Channel 'D' is above 120/125; therefore RPS 'B' actuates. Channel 'G' is above 120/125; therefore RPS 'A' acutates. A full scram results. It's not possible to actuate RPS (at 120/125 scale) without also generating a rod block (at 85/125 scale).

'A' is wrong: Rod withdraw block, only. Plausible to the examinee who doesn't recall the IRM Upscale Trip (RPS) setpoint of 120/125 scale.

'B' is wrong: Rod withdraw block and RPS 'A' half-scram, only . Plausible to the examinee who fails to recognize that the two IRMs that are above 120/125 scale ('D' and 'G') are associated with opposite sides of RPS.

'C' is wrong: Rod withdraw block and RPS 'B' half-scram, only. Plausible for the same reason as for choice 'B'.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 21 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.00	
System ID:	1098544	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LLOT0071	
Topic:	Recall IRM Scrams	
Num Field 1:	3.7	
Num Field 2:	3.6	
Text Field:	215003 A3.03	
Comments:	Level	RO
	Tier	2
	Group	1
	KA # and Rating	215003 A3.03 (3.7/3.6)
	KA Statement	215003 IRM A3.03 Ability to monitor automatic operations of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM including: A3.03 RPS status
	References	ARC-MCR-107, F3, Rev.1 ARC-MCR-107, H3, Rev.0
	Examinee References	None
	Learning Objective	LLOT0071
	Question source	MODIFIED Bank 560453: revised correct answer and revised or replaced distractors to improve plausibility (not possible to get any form of scram (half or full) without also getting a rod block
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.7
	Comments	Minor editorial change to stem and two distractors to address NRC Rev.00 comments.

22 **ID: 1098605** **Points: 1.00**

Unit 1 is shut down.

The PRO is using RCIC in Pressure Control Mode to maintain RPV pressure in an assigned band of 800-900 psig:

- RCIC Flow Controller is in AUTO; its OUTPUT signal is 50%

RPV pressure is 850 psig, steady, when the following occurs:

- RCIC Flow Controller OUTPUT signal goes to 100% (reason unknown)

WHICH ONE of the following describes an **immediate** plant/system response, assuming no operator action?

A. RCIC turbine speed lowers

B. RPV pressure lowers

C. RCIC isolates on high steam line flow

D. CST level rises

Answer: B

Answer Explanation

RCIC is placed in Pressure Control mode using S49.7.A, section 4.3.

Before the Flow Controller failure, RPV pressure is being held constant at 850 psig as a result of the following "balance"...

- RCIC Pump is recircing water to/from the CST at some constant flow rate (gpm)
- Because RCIC water has no other flow path than to/from the CST, the CST level is being held constant at some level (feet)
- A constant controller Output signal of 50% is maintaining the RCIC turbine at some constant speed (rpm)
- A constant amount (Mlbm/hr) of reactor steam is being demanded by the RCIC turbine governor in order to maintain the constant turbine speed
- HV-49-1F022 (RCIC Full Flow Test Valve) is throttled to some constant position, maintaining some constant amount of resistance to RCIC (CST water) flow

When the Flow Controller OUTPUT fails (reason unknown) to 100%, the following results...

- 1F022 position remains unchanged, continuing to resist RCIC (CST water) flow the same amount as before
- The controller sends a significantly higher value (24 mA) signal to the tubine governor system which correspondingly admits more reactor pressure into the steam chest in order to speed up the turbine
- This demand for more reactor steam causes RPV pressure to LOWER
- Of course, the rate (gpm) at which the RCIC Pump recircs CST water is greater than before the failure; however, for the same reason as before, CST level remains unchanged

'B' is correct: RPV pressure lowers. Correct for the reasons described above.

'A' is wrong: RCIC turbine speed lowers. Very plausible to the examinee with a weak comprehension of "Controllers". History has show it is quite common for operators to confuse a "controller failing high" (100% output signal) with, instead, the "feedback sensor (in this case, the RCIC flow element/transmitter) failing high". Were the RCIC flow transmitter fail high (24 mA output signal sent to the flow controller), the controller would "see" a flow that is too high, thereby reducing its (the controller's) output signal which would cause RCIC turbine speed to lower.

'C' is wrong: RCIC isolates on high steam line flow. Plausible to the examinee who recognizes (correctly so) the change in system dynamics that occur when the controller fails high (i.e., that the governor valve opens wide to demand more steam flow be admitted into the steam chest). However, this examinee also believes that such increased steam line flow will reach 300% of rated steam flow (the isolation setpoint); it will not.

'D' is wrong: CST level rises. Plausible to the examinee who recognizes (correctly so) the change in system dynamics that occur when the controller fails high (i.e., that the higher turbine speed translates to a higher RCIC Pump flow, meaning a greater amount of "pump heat" generated within the pump and therefore a heat addition to the CST water; over time, that temperature rise will cause a CST level rise. However, because the stem question uses the qualifier "immediate", this choice is nonetheless wrong.

Question 22 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	2.50	
System ID:	1098605	
User-Defined ID:	REV 02, 01/06/15	
Cross Reference Number:	LLOT0380.11B	
Topic:	Predict RPV pressure response to RCIC flow controlller failure	
Num Field 1:	4.2	
Num Field 2:	4.2	
Text Field:	206000.A1.02	
Comments:	Level	RO
	Tier	2
	Group	1
	KA # and Rating	217000 K3.02 (3.6/3.6)
	KA Statement	217000 RCIC Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on following: K3.02 Reactor vessel pressure
	References	S49.7.A, Rev.11
	Examinee References	None
	Learning Objective	LLOT0380.11B
	Question source	New
	Question history	None
	Cognitive level	Higher

	10 CFR 55 Comments	41.7, 41.10 Major revision to address NRC Rev.00 comments. Minor revision (stem and choice 'A') to address NRC Rev.01 comments.

23 **ID: 1098611** **Points: 1.00**

Unit 1 is operating at 100% power.

D13 DG monthly run is in progress with the D/G running:

- DG frequency is 59.5 Hz
- DG voltage is 4278 Volts with the output breaker OPEN

101 Safeguard Transformer lockout occurs.

CRS directs placing D13 DG in parallel with the 201 Safeguard Bus and loading the DG to 2000 KW.

WHICH ONE of the following identifies the INCOMING source to the D13 Bus and the proper direction of the synchroscope for synchronizing?

	INCOMING SOURCE	SYNCHROSCOPE DIRECTION
A.	201 Bus	Clockwise
B.	201 Bus	Counterclockwise
C.	D13 DG	Clockwise
D.	D13 DG	Counterclockwise

Answer: B

Answer Explanation

With the D13 DG already running at speed when the 101-D13 breaker trips due to the 101 Safeguard Transformer lockout, the D13 DG output breaker will auto close 0.5 seconds after D13 Bus volts drops below 40%. This timing sequence is faster than the Dead Bus Transfer to the 201 source which occurs after a 1 second time delay. With the DG supplying the bus during synchronizing operations, the INCOMING source will be the 201 Bus and the DG will be the RUNNING source. When parelling in this manner, the proper synchroscope operation is in the counter-clockwise direction (SLOW in the "SLOW" direction) since the DG should always operate faster than the offsite source to prevent motoring the DG. Since the synchroscope shows the relative speed difference between the INCOMING compared with the RUNNING, the sycnroscope will show the INCOMING to be running slower which is the counter-clockwise direction. The procedural direction for this is found in S92.2.N, Section 4.5.

'B' is correct: 201 Bus; Counter-clockwise. Correct for the reasons described above.

'A' is wrong: 201 Bus; Clockwise. Plausible to the examinee (a fairly commonly demonstrated weakness) who fails to recongnize the offsite power-to-running DG relationship (i.e., INCOMING-to-RUNNING) described above. Inexperienced examinees are accustomed to thinking in terms of sych scopes rotating "Slow in th Fast" direction; that mindset often works against them when confronted the "reverse" type of relationship depicted in the question.

'C' is wrong: D13 DG; Clockwise. Plausible for similar reasons as that for choice 'A'.

'D' is wrong: D13 DG; Counter-clockwise. Plausible for similar reasons as that for choices 'A' and 'C'.

Question 23 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	2.50
System ID:	1098611
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LGSOPS0092B.IL12
Topic:	Recognize Incoming-Running power source relationship and determine how to parallel those sources
Num Field 1:	3.1
Num Field 2:	3.4
Text Field:	262001 K5.01
Comments:	Level RO Tier 2 Group 1 KA # and Rating 262001 K5.01 (3.1/3.4) KA Statement 262001 AC Electrical Distribution Knowledge of the operational implications of the following concepts as they apply to A.C. ELECTRICAL DISTRIBUTION: K5.01 Principle involved with paralleling two A.C. sources References S92.2.N, Rev.34 Examinee References None Learning Objective LGSOPS0092B.IL12 Question source Bank 742515 Question history None Cognitive level Higher 10 CFR 55 41.5 Comments None

24 **ID: 1109928** **Points: 1.00**

Unit 1 is operating at 100% power.

Operators are preparing to perform ST-6-076-250-1 (SGTS and RERS Flow Test).

WHICH ONE of the following describes the performance of this ST?

A. Uses HVAC ISOLATION 'A' and 'B' pushbuttons to simultaneously start both SGTS trains

B. Starts one SGTS fan at a time with flow through only one Carbon Filter

C. Starts one SGTS fan with both RERS fans at the same time

D. Tests SGTS separately from RERS

Answer: B

Answer Explanation

A review of ST-6-076-250-1 reveals the following:

- Section 4.4 starts only SGTS 'A' fan along with only RERS 'A' fan
- Section 4.5 does the same, except it tests only the 'B' fan for each system
- Sections 4.6 and 4.7 test only one SGTS train at a time, with its suction from the refuel floor

'B' is correct: Starts one SGTS fan at a time with flow through only one Carbon Filter. Correct for the reasons described above.

'A' is wrong: Uses HVAC ISOLATION 'A' and 'B' pushbuttons to simultaneously start both SGTS trains. Plausible to examine who is not familiar with this ST, explicity, and so concludes that the ST would be conducted with SGTS in its "normal" automatic initiation alignment (i.e., with bith SGTS trains in AUTO).

'C' is wrong: Starts one SGTS fan with both RERS fans at the same time. The relationship between SGTS and RERS is an often misunderstood concept among inexperienced licensed operator candidates; plausible for this reason.

'D' is wrong: Tests SGTS separately from RERS. The relationship between SGTS and RERS is an often misunderstood concept among inexperienced licensed operator candidates; plausible for this reason.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 24 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.00	
System ID:	1109928	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	NONE	
Topic:	Recall SGTS Tech Spec Surveilllance Method	
Num Field 1:	3.7	
Num Field 2:	4.1	
Text Field:	261000 2.2.12	
Comments:	Level	RO
	Tier	2
	Group	1
	KA # and Rating	261000 2.2.12 (3.7/4.1)
	KA Statement	261000 SGTS 2.2.12 Knowledge of surveillance procedures.
	References	ST-6-076-250-1, Rev.48
	Examinee References	None
	Learning Objective	No specified objective
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.10
	Comments	Fixed typo in choice 'A' to address NRC Rev.00 comments.

25 **ID: 1110346** **Points: 1.00**

Unit 1 is operating at 100% power when the following occurs:

- Main turbine trip
- An ATWS occurs

RPV pressure rapidly rises.

No SRVs open until RPV pressure reaches 1190 psig.

WHICH ONE of the following identifies the **MINIMUM** number of SRVs that have failed?

A. 4

B. 5

C. 9

D. 10

Answer: C

Answer Explanation

There are 14 SRVs; their "safety" relief pressures are:

1170 psig - SRVs H, J, L, N (a group of 4)
1180 psig - SRVs D, E, K, M, S (a group of 5)
1190 psig - SRVs A, B, C, F, G (a group of 5)

Stem indicates that no SRVs open until pressure reaches 1190 psig; therefore, the first group of 4 have failed (they should have opened at 1170 psig) and the second group of 5 have failed (they should have opened at 1180 psig). All total, a minimum of 9 SRVs failed to open.

'C' is correct: 9. Correct for the reasons described above.

'A' is wrong: 4. This Exam Author has discovered that the SRV lift setpoints and the number of SRVs in each lift setpoint group is a general weakness among ILT Candidates. As such, this choice is plausible to the examinee who incorrectly recalls the setpoints as being (in sequence): 1180, 1190, 1200 psig.

'B' is wrong: 5. Plausible for reasons similar to that for choice 'A'. In this case, the examinee also incorrectly recalls the "order" of the groups; i.e.: 5 + 4 + 4.

'D' is wrong: 10. Plausible for reasons similar to that for choices 'A' and 'B'.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 25 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.50
System ID:	1110346
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LGSOPS0001B.IL2B
Topic:	Recall SRV safety/relief lift setpoints
Num Field 1:	4.2
Num Field 2:	4.4
Text Field:	239002 K3.02
Comments:	Level RO Tier 2 Group 1 KA # and Rating 239002 K3.02 (4.2/4.4) KA Statement 239002 SRVs Knowledge of the effect that a loss or malfunction of the RELIEF/SAFETY VALVES will have on following: K3.02 Reactor over pressurization References LGSOPS0001B, Main Steam / Pressure Relief lesson plan Examinee References None Learning Objective LGSOPS0001B.IL2B Question source New Question history None Cognitive level Lower 10 CFR 55 41.7 Comments None

26 **ID: 1102330** **Points: 1.00**

Unit 2 reactor startup is in progress, with the following:

- All IRMs are on Range 4
- All SRMs are reading 600 CPS
- Rod withdrawals are in progress

An RPV instrument dry tube cracks and fills with water; as a result:

- Interior of SRM 'C' detector fills with water

Delete 026 per 3/10/15

WHICH ONE of the following identifies the SRM 'C' recorder and RMCS Rod Block response?

	SRM 'C' recorder	RMCS Rod Block
A.	Indicates UPSCALE	Yes
B.	Indicates UPSCALE	No
C.	Indicates DOWNSCALE	Yes
D.	Indicates DOWNSCALE	No

Answer: A

Answer Explanation

Per BWR Generic Fundamentals (GFE) lesson plan for "Sensors & Detectors", learning objective #24, the "wetting" of a fission chamber detector causes the detector to fail HIGH (recorder goes UPSCALE). The SRM Upscale (>1 x 10^5 CPS) Rod Block is enabled so long as its associated IRMs are below Range 8.

'A' is correct: Indicates UPSCALE; Yes. Correct for the reasons described above. All IRMs are on Range 4, so the Upscale rod block function is enabled.

'B' is wrong: Indicates UPSCALE; No. Plausible to the examinee who fails to recall the SRM Upscale rod block or who mistakenly believes the Upscale rod block is bypassed with IRMs on Range 4.

'C' is wrong: Indicates DOWNSCALE; Yes. Plausible to the examinee who doesn't recall the GFE knowledge regarding the failure modes for fission chamber detectors. That examinee believes the detector (and therefore the recorder) fails downscale, and as such results in an SRM DOWNSCALE (<3 CPS) Rod Block.

'D' is wrong: Indicates DOWNSCALE; No. First part is plausible for the same reason as for choice 'C'. The second part is plausible to the examinee who either confuses the SRM DOWNSCALE rod block function (which is NOT dependent on the IRM Range) with the SRM NOT FULL INSERTED AND <100 CPS rod block function (which IS bypassed when the IRMs are on Range 3 or higher), or who simply believes that the SRM downscale rod block function is bypassed with the IRMs on Range 4.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 26 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	1102330
User-Defined ID:	REV 01, 12/15/14
Cross Reference Number:	LGSOPS0074.IL6
Topic:	Predict SRM and RMCS response to SRM detector wetting
Num Field 1:	2.9
Num Field 2:	2.9
Text Field:	215004 K6.04
Comments:	Level: RO Tier: 2 Group: 1 KA # and Rating: 215004 K6.04 (2.9/2.9) KA Statement: 215004 SRMs Knowledge of the effect that a loss or malfunction of the following will have on the SOURCE RANGE MONITOR (SRM) SYSTEM: K6.04 Detectors References: ARC-MCR-107, G4, Rev.0 ARC-MCR-107, H4, Rev.1 BWR Fundamentals Chapter 7 (Components), Rev.4 Examinee References: None Learning Objective: LGSOPS0074.IL6 Question source: New Question history: None Cognitive level: Higher* 10 CFR 55: 41.7 Comments: *Justification for HIGHER COGNITIVE categorization: First part requires comprehension of GFE "Components" theory regarding fission chamber "wetting". Minor editorial change to answer choices to address NRC Rev.00 comments.

Delete Q26 per 3/10/15

27 **ID: 1102331** **Points: 1.00**

Unit 1 plant heatup is in progress.

RPV pressure is 50 psig.

A electrical short causes the loss of all power from 1AY160.

WHICH ONE of the following identifies an automatic plant response?

A. Both SGTS Exhaust Fans auto-start

B. RPS half-scram due to APRM INOP

C. RBM 'A' Downscale Trip

D. MAIN STEAM LINE HIGH-HIGH RADIATION alarm

Answer: D

Answer Explanation

Refer to 1S26.1.Q (COL), Alignment of the Main Steam Line Radiation Monitors for Normal Operations, Step 2, which shows that breaker 10AY160-02 feeds power (RPS UPS 120VAC) to "Rad Monitor Instrument Panel 10C606." That panel supplies power to Main Steam Line (MSL) Radiation Monitors 'A' and 'C'. Refer to alarm response card ACR-MCR-107, I1 (Main Steam Line High-High Radiation), specifically Cause #2..."detector inoperable"...this means that a loss of high voltage power (as in the case of a loss of 1AY160 UPS power) to a MSL Rad Monitor detector causes the Rad Monitor Channel to trip UPSCALE.

'D' is correct: MAIN STEAM LINE HIGH-HIGH RADIATION alarm. Correct for the reasons described above.

'A' is wrong: Both SGTS Exhaust Fans auto-start. Refer to Event procedure E-1AY160 (Loss of 1A RPS UPS Power), Confirming Inidication 1.7, which shows that a loss of this UPS causes an auto-start of the 'A' SGTS Fan (only). Notice that Confirming Indications 1.9 and 1.10 show that all Reactor Enclosure Supply and Exhaust Fans trip. This choice is plausible to the examinee who recalls the trip of all R.E. fans and mistakenly translates this to a bondafide RE HVAC "isolation" condition; if it were (it is NOT), then that examinee would conclude that both SGTS trains auto-start.

'B' is wrong: RPS half-scram due to APRM INOP. This choice is plausible to the examinee who confuses RPS UPS power (1AY160, 1BY160) with the 1AD185 (1BD185) UPSs that power the APRM channels. If either of those supplies (1AD185 or 1BD185) were lost, the result would be a trip of 2 of the 4 VOTER units, which would result in an RPS half-scram due to an APRM INOP condition.

'C' is wrong: RBM 'A' Downscale Trip. Plausible to the examinee who fantly recall some association between RPS UPS power and the RBM channels; there is...RPS 'B' UPS power (1BY160) powers the RBM channel recorders on MCR panel 10C603. It is 1AD185 and 1BD185 UPS power that supplies the RBM channels, although RPS UPS power (1AY160, 1BY160) does provide backup power to the RBM channels.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 27 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.00	
System ID:	1102331	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	LGSOPS0026A.IL5B	
Topic:	Predict plant response to loss of 1AY160	
Num Field 1:	2.8	
Num Field 2:	3.0	
Text Field:	262002 K1.14	
Comments:	Level	RO
	Tier	2
	Group	1
	KA # and Rating	262002 K1.14 (2.8/3.0)
	KA Statement	262002 UPS (AC/DC) Knowledge of the physical connections and/or cause-effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the following: K1.14 Main steam line radiation monitors
	References	E-1AY160, Rev.27 1S26.1.Q (COL), Rev.2 ARC-MCR-107, I1, Rev.1 E-1BY160, Rev.23
	Examinee References	None
	Learning Objective	LGSOPS0026A.IL5B
	Question source	New
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.4
	Comments	None

28 **ID: 1102409** **Points: 1.00**

Unit 1 LOCA is in progress, with the following:

- RPV level is -135", down slow
- RPV pressure is 490 psig, down slow
- 1B and 1D RHR Pumps are running

WHICH ONE of the following will provide **DEFINITIVE** evidence that some amount of RHR Loop 'B' water is actually injecting into the RPV?

A. FI51-1R603B (LOOP B) reads 4,000 gpm

B. HV51-1F041D (INBOARD CHECK) "Disc Pos" indicates OPEN, "Stem Pos" indicates CLOSED

C. DIV 2 LPCI INJECTION VALVE DP PERMISSIVE alarm

D. 1B RHR PUMP DISCH HI/LO PRESS alarm

Answer: ~~B~~ KP 3/10/15 Accept A or B

Answer Explanation

Refer to RHR P&ID M-0051, Sheet 3, Coordinates F/G-7, which shows the LPCI 'D' Inboard [Testable] Check Valve HV-1F041D. On MCR panel 10C601, this valve has a remote-position indicator lamp with two halves. The top half of the lamp is labeled "Disc Pos", the bottom half is labeled "Stem Pos". Only by actual flow pushing against the check valve's disc will the "Disc Pos" half of the lamp indicate OPEN (i.e., Green light OFF, Red light ON). The "Stem Pos" half of the lamp will indicate OPEN only when operators apply a TEST signal (TEST pushbutton) to surveill the check valve's operability; in that case, the "Stem Pos" is indicative of the position of the penumatic operator used to open the valve. The LPCI 'D' injection line flow indicator (FI-1R603D) is shown at P&ID Coordinate G-3. This instrument senses flow upstream of the line's injection valve (1F017D); therefore, this indication is NOT definitive evidence of injection into the RPV.

'B' is correct: HV51-1F041D (INBOARD CHECK) "Disc Pos" indicates OPEN, "Stem Pos" indicates CLOSED. Correct for the reasons described above.

'A' is wrong: FI51-1R603B (LOOP B) reads 4,000 gpm. Plausible to the examinee for three primary reasons: 1) at LGS, use of the INBOARD CHECK valve indication is given very ~~little emphasis~~, if any, during the course of the Simulator Phase; 2) students ~~are accumstomed~~ to using the injection line flow indicators to determine ~~when RHR is injecting~~; and 3) many examinees have never given a thought to ~~exactly where~~ in the system the flow element for FI51-1R603B is located. KP 3/10/15 Accept A or B

'C' is wrong: DIV 2 LPCI INJECTION VALVE DP PERMISSIVE alarm. Refer to alarm resposne card ARC-MCR-115, F4 (DIV 2 LPCI INJECTION VALVE DP PERMISSIVE). This alarm simply alerts operators to the fact that all conditions are satisfied for the automatic opening of the LPCI 'B' injection valve (F017B). Under expected LOCA conditions, this alarm is expected at an RPV pressure well above the RHR pump shutoff head pressure (~270 psig). Plausible to the examinee who confuses the opening of the injection valve with conditions that would allow actual injection flow into the RPV.

'D' is wrong: 1B RHR PUMP DISCH HI/LO PRESS alarm. Refer to alarm response card ARC-MCR-115, F3. This is NOT an expected alarm so long as the RHR pump has an unobstructed flowpath at its discharge. Plausible to the weaker examinee who believes the alarm to be indicative of sufficent pump discharge pressure so as to cause sufficient injection flow into the RPV.

Question 28 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.00	
System ID:	1102409	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	LGSOPS0051	
Topic:	Recognize definitive MCR indication of LPCI injection flow into the RPV	
Num Field 1:	2.7	
Num Field 2:	2.9	
Text Field:	203000 K5.01	
Comments:	Level	RO
	Tier	2
	Group	1
	KA # and Rating	203000 K5.01 (2.7/2.9)
	KA Statement	203000 RHR/LPCI: Injection Mode Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI: INJECTION MODE (PLANT SPECIFIC): K5.01 Testable check valve operation
	References	P&ID M-0051, Sheet 3, Rev.68 ARC-MCR-115, F4, Rev.0 ARC-MCR-115, F3, Rev.1
	Examinee References	None
	Learning Objective	LGSOPS0051, no specified objective
	Question source	New
	Question history	None

	Cognitive level	Higher
	10 CFR 55	41.5
	Comments	None

29 **ID: 1102424** **Points: 1.00**

Unit 2 is operating at 100% power.

The following occurs:

- FWLCS TROUBLE alarm is received
- The alarm is due to the control signal failing to 0 mA on RFPT B
- No operator action is taken

Then:

- A main turbine trip occurs
- An automatic RPS scram is successful

WHICH ONE of the following identifies:

(1) the operating mode of RFPT B (M/A or MSC), and
(2) whether or not the Scram Profile activated (Yes/No)?

A. (1) MSC
(2) No

B. (1) MSC
(2) Yes

C. (1) M/A
(2) Yes

D. (1) M/A
(2) No

Answer: B

Answer Explanation

Prior to the control signal failure on RFPT B, the control mode for all three RFPTs (A, B, C) is "M/A" (Master Automatic). The essential information to determine how RFPT B responds when its control signal is completely lost (i.e., 0 mA signal) is found in S06.1.H (Responding to Alarms and Selected Events at the Feedwater Level Control System Operator Station), specifically, Attachment 1 Alarm List, page 17, signal identity 2XX-FW311.ICSF. There, we find that a complete control signal failure ("outside the 4-20 mA range") automatically transfers the affected RFPT control to the "MSC" (Master Speed Change) control mode and locks up that RFPT at its current speed. With "no operator action taken" that remains the RFPT B control mode both prior to and after the RPS scram. The successful RPS scram (i.e., de-energization of the K14 "scram relays" in both RPS Trip Systems (A and B) results in the automatic activation of the FWLCS "Scram Profile" (so long as Three-Element level control is available). The "locked up" RFPT B has no impact on Scram Profile activation.

'B' is correct: (1) MSC; (2) Yes. Correct for the reasons described above.

'A' is wrong: (1) MSC; (2) No. Plausible to the examinee who does recognize how RFPT B responds to the control signal failure, but who incorrectly concludes that, along with the availability of Three-Element Level Control, all 3 RFPTs must also be operating in M/A control mode in order for the Scram Profile to activiate.

'C' is wrong: (1) M/A; (2) Yes. Plausible to the examinee who incorrectly recalls the information found in the Alarm List discussed above. There, we find that, unlike the case of a "hard" failure of the control signal (i.e., outside the 4-20 mA range), a "soft" failure (i.e., a control signal deviation condition) does NOT swap the RFPT to MSC control mode, thus leavnig RFPT B operating in M/A mode.

'D' is wrong: (1) M/A; (2) No. The first part is plausible for the same reason as for choice 'C'. The second part is plausible to the examinee who inccoretly concludes that a single RFPT "control signal failure" behaves much like, for example, a TOTAL FW FLOW ERROR condition...refer to the Alarm List, signal identity 2XX-FW302.ITFFE, where we see that such a "failure" condition will in fact disable the Scram Profile.

Question 29 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.50
System ID:	1102424
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LLOT0550.4
Topic:	Predict FWLCS response to a single RFPT control signal loss (hard failure)
Num Field 1:	3.1
Num Field 2:	3.2
Text Field:	259002 K4.06
Comments:	Level: RO Tier: 2 Group: 1 KA # and Rating: 259002 K4.06 (3.1/3.2) KA Statement: 259002 Reactor Water Level Control Knowledge of REACTOR WATER LEVEL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: K4.06 Control signal failure References: S06.1.H U/2, Rev. 8 ARC-MCR-107, D5, Rev.5 Examinee References: None Learning Objective: LLOT0550.4 Question source: New Question history: None Cognitive level: Lower 10 CFR 55: 41.7 Comments: None

30 **ID: 1102426** **Points: 1.00**

Unit 1 operators are ready to commence a GP-2, Appendix 1 reactor startup.

Initial conditions:

- All control rods fully inserted
- All SRMs are reading 100 CPS

Following the withdrawal of 20 control rods, all SRMs are reading 200 CPS.

WHICH ONE of the following completes the statement below?

As criticality is approached, GP-2 Appendix 1 requires that the operator use only single-notch rod motion between __(1)__ until the reactor is critical and one turbine bypass valve is open. Following the initial withdrawal of the 20 control rods (and assuming equal rod worth for all rod withdrawals) an additional __(2)__ must be withdrawn to reach criticality.

A. (1) positions 04 and 36
(2) 40 rods

B. (1) positions 12 and 20
(2) 20 rods

C. (1) positions 04 and 36
(2) 20 rods

D. (1) positions 12 and 20
(2) 40 rods

Answer: C

Answer Explanation

Refer to GP-2, Appendix 1, steps 3.2.9 and 3.2.13. When three SRM count rate doublings is reached, step 3.2.13.4 directs operators to use only single-notch rod withdrawals between notches 04 and 36 until the reactor is critical and one turbine bypass valve is open. Refer to BWR Reactor Theory Chapter 8, "Reactor Operational Physics", pages 13-28, regarding "subcritical multiplication" and the approach to reactor criticality. From that "Generic Fundamentals" training, we recall that the distance to criticality is halved with each additional, equal amount of reactivity (% delta-K/K). One essential point to conclude from this is...with each halving of the distance to criticality, the SRM count rate doubles. The GFE "rule of thumb" is..."When enough reactivity is added to the reactor to double the count rate, if the same reactivity is added to the reactor again, the reactor will be supercritical."

'C' is correct: (1) positions 04 and 36; (2) 20 rods. Part (1) is taken directly from step 3.2.13.4 of GP-2, Appendix 1. Part (2) is correct based on the stem condition fact that 20 rods produced the first count rate doubling; therefore, another 20 rods (assuming equal rod worths) will bring the reactor critical.

'A' is wrong: (1) positions 04 and 36; (2) 40 rods. Part (2) is plausible to the examinee incorrectly recalls the GFE thumb rule. Instead, that examinee believes that if "double" (i.e., twice) the amount of the initial rod worth (of 20 rods) is added, the reactor will reach criticality. As such, the examinee concludes that an additional 40 rods (i.e., 2 x the initial 20) are withdrawn, criticality will be reached.

'B' is wrong: (1) positions 12 and 20; (2) 20 rods. Part (1) is plausible to the examinee who cannot recall the GP-2, Appendix 1, step 3.2.13.4 requirement. However, that examinee does recall the generalilty concerning "integral rod worth" as described in Generic Fundamentals Reactor Theory Chapter 5 (Control Rods), page 16 and Figure 5-6, where we discover that the highest integral rod worth is between notches 12 and 16; thus, this choice (modified to read "between positions 12 and 20") provides that distraction.

'D' is wrong: (1) positions 12 and 20; (2) 40 rods. Plausible for the same reasons as that for choices 'A' and 'B'.

Question 30 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.50
System ID:	1102426
User-Defined ID:	REV 01, 12/15/14
Cross Reference Number:	
Topic:	Recall GP-2 Appx 1 guidance regarding SRM count rate doublings
Num Field 1:	4.4
Num Field 2:	4.7
Text Field:	2.1.7
Comments:	Level: RO Tier: 3 Group: N/A KA # and Rating: 2.1.7 (4.4/4/7) KA Statement: 2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. References: GP-2, Appendix 1, Rev.49 GFE Reactor Theory Chapter 5, Rev.4 GFE Reactor Theory Chapter 8, Rev.4 Examinee References: None Learning Objective: No specified objective Question source: New Question history: None Cognitive level: Higher 10 CFR 55: 41.5 Comments: Editorial changes to stem and answer choices to address NRC Rev.00 comments.

31 **ID: 1102445** **Points: 1.00**

Tech Spec LCO 3.0.3 is entered when an LCO applicable in OPCON 1 cannot be met.

WHICH ONE of the following states the **MAXIMUM** allowed time to perform the specified action?

A. Must begin reducing power upon entry into the LCO

B. Must be in OPCON 3 within 6 hours

C. Must be in OPCON 3 within 13 hours

D. Must be in OPCON 4 within 24 hours

Answer: C

Answer Explanation

Tech Spec LCO 3.0.3 reads...

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in an OPERATIONAL CONDITION in which the Specification does not apply by placing it, as applicable, in:

a. At least STARTUP within the next 6 hours.
b. At least HOT SHUTDOWN within the following 6 hours, and
c. At least COLD SHUTDOWN within the subsequent 24 hours.

'C' is correct: Must be in OPCON 3 within 13 hours. OPCON 3 is HOT SHUTDOWN. Operator have a MAXIMUM of 13 hours, total (from the time of entering the LCO) to place the plant in HOT SHUTDOWN. The 13 hours considers: 1 hour to take action + 6 hours thereafter to be in STARTUP (OPCON 2) + 6 hours thereafter to be in HOT SHUTDOWN (OPCON 3), for a total of 13 hours.

'A' is wrong: Must begin reducing upon entry into the LCO. Plausible to the examinee who simply cannot recall the "1 hour to take action" requirement.

'B' is wrong: Must be in OPCON 3 within 6 hours. Plausible to the examinee who forgets that the time requirements are additive, but instead only recalls the "At least HOT SHUTDOWN within the following 6 hours" requirement of the LCO.

'D' is wrong: Must be in OPCON 4 within 24 hours. Plausible for reasons similar to that for choice 'B'; in this case, the examinee forgets the time requirements are additive and recall the "At least COLD SHUTDOWN within the following 24 hours" requirement of the LCO.

NOTE - Exam Author, Facility Rep, and Technical Reviewer all agree that LCO 3.0.3 in its entirety (including its ACTIONs and ACTION TIMEs) is RO-level knowledge.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 31 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.00	
System ID:	1102445	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LGSOPS1800.5	
Topic:	Recall LCO 3.0.3 requirements	
Num Field 1:	4.0	
Num Field 2:	4.7	
Text Field:	2.2.22	
Comments:	Level	RO
	Tier	3
	Group	N/A
	KA # and Rating	2.2.22 (4.0/4.7)
	KA Statement	Knowledge of limiting conditions for operations and safety limits.
	References	Tech Spec LCO 3.0.3 (latest)
	Examinee References	None
	Learning Objective	LGSOPS1800.5
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.5
	Comments	Major revision to address NRC Rev.00 comments.

32 **ID: 1117935** **Points: 1.00**

Unit 1 operators are venting Primary Containment using T-200 (Primary Containment Emergency Vent Procedure).

WHICH ONE of the following would require the operators to manually isolate the vent pathway?

A. Venting via the 2" Suppression Pool Vent when a SOUTH STACK HI HI RADIATION alarm is received

B. Venting via the 18" Suppression Pool Purge Exhaust when a NORTH STACK HI HI RADIATION alarm is received

C. Venting via the 24" Drywell Purge Exhaust when a NORTH STACK WRAM HI HI RADIATION alarm is received

D. Venting via the 24" Drywell Purge Supply when a SOUTH STACK HI HI alarm is received

Answer: A

Answer Explanation

Refer to U/1 T-200:

Section 4.2 is used to vent via the 2" suppression pool vent; CAUTION #2 as well step 4.2.6 directs operator to manually isolate this release pathway should the South Stack hi Hi Radiation alarm be received.

Section 4.3 is used to vent via the 18" suppression pool purge exhuast; CAUTION #2 reminds the operators that this release pathway will automatically isolate should the North Stack Hi Hi Radiation alamrm be received.

Section 4.5 is used to vent via the 24" drywell purge exhaust; CAUTION #2 reminds the operators that this release pathway will automatically isolate should the North Stack WRAM Hi Hi alarm be received.

Section 4.8 is used to vent via the 24" drywell purge supply; CAUTION #2 reminds operators that this release pathway is used only if venting with exceeding offsite release limits is necessary (i.e., there is no requirement to isolate this pathway because it's already understood that venting the PC has a higher a priority than does controlling the offsite release).

'A' is correct: Venting via the 2" Suppression Pool Vent when a SOUTH STACK HI HI RADIATION alarm is received. Correct for the reasons described above.

'B' is wrong: Venting via the 18" Suppression Pool Purge Exhaust when a NORTH STACK HI HI RADIATION alarm is received. Plausible to the examinee who fails to recall that this release pathway will automatically isolate in response to this alarm.

'C' is wrong: Venting via the 24" Drywell Purge Exhaust when a NORTH STACK WRAM HI HI RADIATION alarm is received. Plausible to the examinee who fails to recall that this release pathway will automatically isolate in response to this alarm.

'D' is wrong: Venting via the 24" Drywell Purge Supply when a SOUTH STACK HI HI alarm is received. Very plausible due to the fact that it suggests the South Stack is alarming; this examinee recalls that there is no auotmatic isolation associated with it, therefore, he/she is quick to beleive that a manual isolation of the pathway is directed.

Question 32 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.50	
System ID:	1117935	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LGSOPS2003.IL3	
Topic:	Recognize condition requiring manualy isolation of PC vent pathway	
Num Field 1:	3.5	
Num Field 2:	4.3	
Text Field:	2.3.11	
Comments:	Level	RO
	Tier	3
	Group	N/A
	KA # and Rating	2.3.11 (3.8/4.3)
	KA Statement	2.3.11 Ability to control radiation releases.
	References	T-200 U/1, Rev.24
	Examinee References	None
	Learning Objective	LGSOPS2003.IL3
	Question source	New
	Question history	None
	Cognitive level	Lower*
	10 CFR 55	41.10, 4.11
	Comments	*Replacement question (overlap with Op Test) to address NRC Rev.00 comments. Previous question was categorized as Higher Cognitive.

33 **ID: 1102449** **Points: 1.00**

Unit 1 is operating at 100% reactor power when the following MCR alarms are received:

- MOTOR DRIVEN FIRE PUMP AUTO START
- REAC 1 EL 177 PB NE STAIR (on fire panel)

WHICH ONE of the following identifies an Immediate Operator Action of SE-8 (Fire)?

A. Notify Security to escort offsite fire departments to location requested by Fire Brigade Leader

B. Start the Diesel Driven Fire Pump

C. Evacuate Unit 1 Reactor Enclosure elevations 177', 217', and 253'

D. Record the time and inform the Shift Manager

Answer: D

Answer Explanation

Refer to SE-8, step 3.2. Stem indicates there is in fact an actual fire (fire alarm + fire pump auto-start); therefore, step 3.2.3 applies.

'D' is correct: Record the time and inform the Shift Manager. Correct for the reason described above.

'A' is wrong: Notify Security to escort offsite fire departments to location requested by Fire Brigade Leader. Plausible to the examinee who cannot recall the SE-8 IOAs but who does recall that notifying Security is required by SE-8, Attachment 8, step 1.2; i.e., this is NOT an Immediate Operator Action (IOA).

'B' is wrong: Start the Diesel Driven Fire Pump. This action is called for in Follow-up Action Step 4.6.2.1; it is NOT an IOA. Plausible to the examinee who does recall this action but mistakenly believes it to be an IOA.

'C' is wrong: Evacuate Unit 1 Reactor Enclosure elevations 177', 217', and 253'. While IOA step 3.2.2 does direct operators to execute the evacuation/announcement requirements of Attachment 8, the Attachment 8, itself, does NOT require that any more than the location of the fire (at most, only R.E. Elevation 177', in the case of this question) be evacuated. Plausible to the examinee who recall the IOA that directs evacuations, but who does not recall that the evacuation need only be limited to the location of the fire.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 33 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.00	
System ID:	1102449	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LLOT1563.02	
Topic:	SE-8 - Recall Immediate Operator Actions for Fire In Plant	
Num Field 1:	3.4	
Num Field 2:	3.9	
Text Field:	2.4.27	
Comments:	Level	RO
	Tier	3
	Group	N/A
	KA # and Rating	2.4.27 (3.4/3.9)
	KA Statement	2.4.27 Knowledge of "fire in the plant" procedures.
	References	SE-8, Rev.50
	Examinee References	None
	Learning Objective	LLOT1563.02 (in lesson plan LGSOPS2000)
	Question source	Bank 973779
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.10
	Comments	Major revision to address NRC Rev.00 comments.

34 **ID: 1102450** **Points: 1.00**

WHICH ONE of the following events would, by itself, require an entry into T-103 (Secondary Containment Control)?

A. 1 REAC ENCL REFUEL FLR VENT EXHAUST RAD MON A/B HI-HI / DOWNSCALE alarm due to confirmed valid R.E. exhaust rad hi-hi

B. CONT. ENCL STEAM FLOODING DAMPER PNL 00C728 TROUBLE alarm; damper closure is confirmed

C. REFUELING FLOOR LOW DELTA P alarm; d/p is confirmed to be -0.1" w.g., steady

D. REACTOR ENCL AREA HI RADIATION alarm due to confirmed valid alarm on the SLC System Area Radiation Monitor (ARM)

Answer: A

Answer Explanation

Refer to alarm response card ARC-MCR-109, E1 (1 REAC ENCL REFUEL FLR VENT EXHAUST RAD MON A/B HI-HI / DOWNSCALE). Operator Action #2 directs entry into T-103.

'A' is correct: 1 REAC ENCL REFUEL FLR VENT EXHAUST RAD MON A/B HI-HI / DOWNSCALE alarm due to confirmed valid R.E. exhaust rad hi-hi. Correct for the reason described above.

'B' is wrong: CONT. ENCL STEAM FLOODING DAMPER PNL 00C728 TROUBLE alarm; damper closure is confirmed. Alarm response card ARC-MCR-002, C5 provides direction for this alarm. Plausible to the examinee who too quickly relates this alarm to the similar T-103 entry condition..."R.E. steam flooding damper actuation". The Control Enclosure is NOT associated with/connected to the Reactor Enclosure.

'C' is wrong: REFUELING FLOOR LOW DELTA P alarm; d/p is confirmed to be -0.1" w.g., steady. The strongest of the distracters; plausible to the examinee who recalls that a sustained (50 minute time delayed) low d/p (setpoint = -0.1" w.g.) is in fact an automatic R.E. HVAC isolation, and so believes that, by itself, warrants a T-103 entry. It does not; only an R.E. HVAC isolation due to hi-hi radiation is a T-103 entry.

'D' is wrong: REACTOR ENCL AREA HI RADIATION alarm due to confirmed valid alarm on the SLC System Area Radiation Monitor (ARM). A very strong distracter; plausible to the examinee who recalls that any alarming ARM for one of the Areas listed on Table SCC-1 of T-103 is a T-103 entry condition, but who incorrectly concludes that the SLC System Area is one of those Table SCC-1 Areas...it is NOT.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 34 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	2.50
System ID:	1102450
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LLOT1560.2D
Topic:	Recognize T-103 entry condition
Num Field 1:	4.6
Num Field 2:	4.8
Text Field:	295034 2.4.1
Comments:	Level: RO Tier: 1 Group: 2 KA # and Rating: 295034 2.4.1 (4.6/4.8) KA Statement: 295034 Secondary Containment Ventilation High Radiation 2.4.1 Knowledge of EOP entry conditions and immediate action steps. References: T-103, Rev.20 ARC-MCR-109, E1, Rev.2 ARC-MCR-002, C5, Rev.0 ARC-MCR-002, F2, Rev.4 ARC-MCR-109, B4, Rev.2 Examinee References: None Learning Objective: LLOT1560.2D Question source: New Question history: None Cognitive level: Lower 10 CFR 55: 41.10 Comments: None

35 **ID: 1102542** **Points: 1.00**

A bus fault causes the complete loss of all Unit 1 Div 1 125 VDC power.

WHICH ONE of the following describes an impact of this power loss?

A. If RCIC is running, its turbine governor valve goes fully closed

B. Closure of HV-44-1F004 (RWCU Otbd Isol) causes trip of running RWCU pump

C. None of the non-ADS SRVs can be electrically opened

D. 'C' SLC Pump cannot be started from the MCR

Answer: C

Answer Explanation

The non-ADS SRVs have control switches only at MCR panel 10C626 and at the Remote Shutdown Panel (RSP), both of which control only the Div 1 DC-powered solenoid for each fo those SRVs. Therefore, with the loss of all Div 1 125 VDC, none of the non-ADS SRVs can be electrically opened. [Refer to E-1FA, Loss of Division 1 Safeguard 125/250V DC Bus 1FA, Section 1.1 NOTE #3, to validate this.]

'C' is correct: None of the non-ADS SRVs can be electrically opened. Correct for the reason described above.

'A' is wrong: If RCIC is running, its turbine governor valve goes fully closed. Just the opposite happens...the loss of DC results in the RCIC flow controller output signal going fully upscale, causing the turbine to quickly ramp up and trip on mechanical overpseed.

'B' is wrong: Closure of HV-44-1F004 (RWCU Otbd Isol) causes trip of running RWCU pump. Refer to E-1FA, Confirming Inidication 1.1.5, which shows that the RWCU Inboard Isolation Valve, HV-44-1F001, closes on the Div 1 DC buis loss (not the Outboard Isolation Valve, HV-44-1F004). The 1F001 valve closes due to an NSSSS Group 3 half-isolation resulting fromthe bus loss; it trips the 'A' Channel which closes the 1F001 valve. Plausible to the examinee who confuses the "Inboard/Outboard versus Div 1/Div2" association...an often illustrated problem among ILT Candidates.

'D' is wrong: 'C' SLC Pump cannot be started from the MCR. The SLC Pumps are not reliant on DC power; they are 480 VAC MCC-powered motors and control power is supplied from the MCC breaker's internal 480/120 VAC step-down transformer. If required, the 'C' SLC pump will start from its control switch at MCR panel 10C603. However, if necessary to start the 'A' SLC Pump, it will NOT start. This is because Div 1 DC powers a Level Transmitter (LT) in the SLC Storage Tank; losing that DC power causes the LT to fail downscale, resulting in a "preceived" (though not actual) low level in the tank...this low level signal would immediately trip the 'A' SCL pump is ot were already running and prevent its start (manual or automatic) if it were required. Plausible to examinee who cannot accurately recall this rather complicated association between DC power / SLC Tank level transmitters / SLC pumps.

Question 35 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	3.00	
System ID:	1102542	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LGSOPS0001B.IL3J	
Topic:	Determine impact of Unit 1 Div 1 DC Bus loss	
Num Field 1:	3.3	
Num Field 2:	3.3	
Text Field:	295004 AK2.03	
Comments:	Level	RO
	Tier	1
	Group	1
	KA # and Rating	295004 AK2.03 (3.3/3.3)
	KA Statement	295004 Partial or Total Loss of DC Power Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: AK2.03 D.C. bus loads
	References	E-1FA, Rev.11
	Examinee References	None
	Learning Objective	LGSOPS0001B.IL3j
	Question source	New
	Question history	None
	Cognitive level	Higher*
	10 CFR 55	41.7

	Comments	*Justification for Higher Cognitive categorization: Where it might be argued that directly choosing the correct answer requires only recalling two pieces of information (i.e., 1) the locations at which certain of the SRVs can be operated, and 2) the DC Bus required in order to operate those SRVs at each location), a good amount of higher cognitive thinking is required in order to eliminate the 'A' and 'D' distracters, especially. The explanations for each of those distracters speaks for itself in this regard. Based on all of this, the Exam Author believes that an HCL categorization is appropriate in accordance with the discussion found in NUREG-1021, Appendix A, Section C.3.c, 1st paragraph at the top of page 7 of 11. Editorial changes to two answer choices to avoid overlap with Question #22 and to provide symmetry of length, to address NRC Rev.00 comments.

36 **ID: 1102543** **Points: 1.00**

Unit 2 is in OPCON 4 when the following occurs:

- Shutdown Cooling (SDC) is lost
- All forced reactor coolant circulation is lost

Operators raise RPV water level to +90" on UPSET RANGE to establish natural circulation.

An in-progress OPDRV causes RPV water level to slowly lower uncontrollably.

WHICH ONE of the following identifies when natural circulation will be considered to be lost if RPV level continues to lower?

As soon as UPSET RANGE indicated level drops below...

A. +78"

B. +68"

C. +60"

D. +50"

Answer: A

Answer Explanation

Refer to GP-6.2 (Shutdown Operations - Shutdown Condition Tech Spec Actions), Section 3.0 CAUTION. There, we find that natural circulation occurs when level is raised to either +60" on level indicator LI-42-*R605 (which is the SHUTDOWN RANGE level instrument) or to +78" on level indicator LI-42-*R608 (which is the UPSET RANGE level instrument).

'A' is correct: +78". Correct for the reasons descrbed above.

'B' is wrong: +68". Although this level has no specific relevance at LGS, its value is as plausible as is +78", given two considerations: 1) inexperienced ILT Candidates have little or no actual plant operating experience involving the establishment of natural circulation, and 2) the same can be said for their exposure to the concept as part of their simulator phase of training/qualification.

'C' is wrong: +60". This choice would be correct if the stem condition/question statement considered the SHUTDOWN RANGE level instrument, rather than the UPSET RANGE. Plausible to the examinee who cannot recall which of the two level instruments this +60" value applies to.

'D' is wrong: +50". This choice is plausible for reasons that parallel those of choice 'B'.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 36 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.50	
System ID:	1102543	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	LLOT2001A	
Topic:	Recall RPV level associated with natural circulation	
Num Field 1:	3.8	
Num Field 2:	4.1	
Text Field:	295031 EK1.02	
Comments:	Level	RO
	Tier	1
	Group	1
	KA # and Rating	295031 EK1.02 (3.8/4.1)
	KA Statement	295031 Reactor Low Water Level Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: EK1.02 Natural circulation: Plant-Specific
	References	GP-6.2, Rev.51
	Examinee References	None
	Learning Objective	LLOT2001A, no specified objective
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.8, 41.10
	Comments	None

37 **ID: 1102564** **Points: 1.00**

Unit 2 operators have just reduced power, using Reactor Recirc, to stabilize a degrading Main Condenser vacuum.

Plant conditions are now stabilized, with the following:

- Simulated Thermal Power is 70%
- Core Flow is 50 Mlbm/hr
- Recirc Drive Flow is 37,000 gpm

The OPRM ODA's are continuously displaying the following information for the Period Based Detection Algorithm:

	Amplitude	Counts
OPRM 1	1.10	11
OPRM 2	1.11	13
OPRM 3	1.09	15
OPRM 4	1.11	14

WHICH ONE of the following describes the required operator action and the reason for that action?

A. Immediately scram the reactor because the plant is operating in the EXCLUSION REGION

B. Continue to reduce reactor power by inserting control rods to increase the margin of safety to the OPRM UPSCALE TRIP setpoint

C. Immediately scram the reactor because an OPRM UPSCALE TRIP should have occurred

D. Contact I&C to investigate why the OPRM ODAs have activated

Answer: B

Answer Explanation

All three of the following conditions must be present for OPRM channel trips to be enabled:

- Reactor Power (STP) ≥ 29.5%, **and**
- Total Recirc Drive Flow < 60% (of 88,000 gpm), **and**
- Reactor Mode Switch in "RUN" position

Therefore, the stem conditions indicate that the OPRMs are in fact enabled.

When enabled, the OPRM ODAs activate; i.e., start displaying. Stem conditions indicate that the PBDA Amplitude / Consecutive Counts combination, for all OPRM channels, is below the OPRM UPSCALE TRIP value of 1.12 / 14. Therefore, the ODAs, by definition, are simply displaying the OPRM "Pre-Trip" (alarm) values...i.e., NO OPRM scram should have occurred.

'B' is correct: Continue to reduce reactor power by inserting control rods to increase the margin of safety to the OPRM UPSCALE TRIP setpoint. This is the action, specifically..."Operator Actions" #4 and #5...directed by the alarm response card for the OPRM PRE-TRIP alarm (ARC-MCR-108, A3). Note - This answer choice does NOT require an examine, in a closed-book setting, to have memorized ARC actions. Its selection is based solely on the examinee recognizing that inserting rods to reduce power, thereby increasing the margin of safety to the OPRM UPSCALE TRIP setpoint, is the only reasonable action to take.

'A' is wrong: Immediately scram the reactor because the plant is operating in the EXCLUSION REGION. Plausible to the examinee who fails to recognize that no such EXCLUSION REGION exists on the P/F Map when the OPRMs are OPERABLE (which they clearly are). Refer to GP-5 (Steady State Operations), specifically the four versions of the P/F Map shown on Attachment 3, Attachment 4, and Attachment 8 (pages 6 and 7) to validate the fact that only when the OPRMs are inoperable (as illustrated on the two maps of Attachment 8) does an Exclusion Region exist. This examinee is distracted by the fact that the stem conditions of 70% power, with 50 Mlbm/hr core flow, would place the plant solidly within the EXCLUSION zone, if it existed (it does not).

'C' is wrong: Immediately scram the reactor because an OPRM UPSCALE TRIP should have occurred. Plausible to the examinee who cannot interpret the OPRM ODA indications; he/she incorrectly concludes that the values displayed are at or above the actual OPRM Trip (i.e., scram) setpoints...they are NOT.

'D' is wrong: Contact I&C to investigate why the OPRM ODAs have activated. Plausible to the examinee who cannot recall the activation point for the OPRM ODAs.

Question 37 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	6	
Difficulty:	2.50	
System ID:	1102564	
User-Defined ID:	REV 02, 01/06/15	
Cross Reference Number:	LLOT0074A.16	
Topic:	OPRM ODA Indications - Interpret and determine operator action	
Num Field 1:	3.3	
Num Field 2:	3.3	
Text Field:	215005 A3.03	
Comments:	Level	RO
	Tier	2
	Group	1
	KA # and Rating	215005 A3.03 (3.3/3.3)
	KA Statement	215005 APRM / LPRM Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including: A3.03 Meters and recorders
	References	ARC-MCR-108, A3, Rev.3 GP-5, Rev.168

	Examinee	None
	References	
	Learning Objective	LLOT0074A.16
	Question source	Bank 989725
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.7, 41.10
	Comments	Minor editorial changes to stem and answer choices to address NRC Rev.00 comments. Minor revision (stem) to address NRC Rev.01 comments.

38 **ID: 1102594** **Points: 1.00**

Unit 1 is operating at 100% power when the following occurs:

- Main turbine trips
- ATWS occurs
- RPV pressure peaks at 1170 psig

WHICH ONE of the following identifies the automatic response of the ATWS/ARI and ATWS/RPT functions?

	ARI	RPT
A.	Actuates immediately	Actuates after 9 seconds
B.	Actuates immediately	Actuates immediately
C.	Actuates after 9 seconds	Actuates after 9 seconds
D.	Actuates after 9 seconds	Actuates immediately

Answer: B

Answer Explanation

The following are excerpted from the RRCs lesson plan (LGSOPS0036A, Rev.3, page 4)...

The RRCS logic monitors reactor dome pressure and water level. The logic will cause the immediate energization of the ARI valves when either the reactor high pressure trip setpoint or low water level 2 setpoint is reached, or the manual push buttons are armed and depressed. Energization of the RRCS ARI valves depressurizes the scram air header independent of the logic and vent valves of the RPS system. The valves are sized to allow insertion of all control rods to begin within 15 seconds. Additional immediate RRCS response to the initiation signals include Recirculation System pump motor breaker trip immediately if reactor high pressure is received or 9 seconds after a low water level 2 signal is received.

The ARI function will be initiated by any of the following conditions:

1) High RPV pressure (1149 psig), or
2) Low RPV level (-38 inches), or
3) Manual initiation from MCR

The ATWS/RPT function will be initiated by either of the following conditions:

1) High RPV pressure (1149 psig) (sealed-in), or
2) Low RPV level (-38 inches with a 9-second time delay) (sealed-in)

'B' is correct: Actuates immediately; Actuates immediately. Correct for the reasons described above.

'A', 'C', 'D' are wrong: Wrong for the reasons decribed above. Each is plausible to the examinee whose recall cannot distinguish between the several variations of the RRCS logic described above.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 38 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.00	
System ID:	1102594	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	LGSOPS0036A.3	
Topic:	Recall ATWS/ARI and RPT Functions	
Num Field 1:	4.1	
Num Field 2:	4.2	
Text Field:	295037 EK2.03	
Comments:	Level	RO
	Tier	1
	Group	1
	KA # and Rating	295037 EK2.03 (4.1/4.2)
	KA Statement	295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following: EK2.03 ARI/RPT/ATWS: Plant-Specific
	References	LGSOPS0036A, RRCS lesson plan
	Examinee References	None
	Learning Objective	LGSOPS0036A.3
	Question source	Bank 561532

	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.7
	Comments	None

39 **ID: 1102599** **Points: 1.00**

WHICH ONE of the following describes the reason for performing an emergency blowdown in response to high drywell temperature, per the T-102 (Primary Containment Control) Bases?

A. Maintain ADS SRV availability

B. Prevent exceeding the Tech Spec LCO limit for drywell temperature

C. Prevent RPV level instrument reference leg boiling

D. Prevent exceeding the containment Heat Capacity Temperature Limit (HCTL)

Answer: A

Answer Explanation

The complete explanation (with respect to the context of this question) is found on pages 128 thru 130 of the T-102 Bases, specifically regarding Steps DW/T-10 thru 13. All of that discussion reduces to the following: 340°F is the maximum temperature at which the ADS SRVs are environmentally qualified; it is also the Drywell design temperature.

'A' is correct: Maintain ADS SRV availability. Correct for the reasons described above.

'B' is wrong: Prevent exceeding the Tech Spec LCO limit for drywell temperature. The Tech Spec LCO 3.6.1.7 Drywell Temperature limit is 145°F. Plausible to the examinee who forgets that the drywell temperature is already well above 145°F by the time operators arrive at Step DW/T-10.

'C' is wrong: Prevent RPV level instrument reference leg boiling. Plausible to the examnee who doesn't recall the basis but who does recall that elevated drywell temperatures can lead to RPV level instrument inaccuracies due to reference leg boiling.

'D' is wrong: Prevent exceeding the containment Heat Capacity Temperature Limit (HCTL). Plausible to the examinee who confuses the suppression pool temperature focus of HCTL with elevated drywell temperatures.

NOTE - At LGS, recall of EOP bases is RO-level knowledge supported by learning objectives.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 39 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.00	
System ID:	1102599	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LLOT1560.5	
Topic:	T-102, Recall basis for step DW/T-10 (Blowdown when cannot maintain below 340F)	
Num Field 1:	3.6	
Num Field 2:	397	
Text Field:	295028 EK3.01	
Comments:	Level	RO
	Tier	1
	Group	1
	KA # and Rating	295028 EK3.01 (3.6/3.9)
	KA Statement	295028 High Drywell Temperature Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE: EK3.01 Emergency depressurization
	References	T-102, Rev.24 T-102 Bases, Rev.24
	Examinee References	None
	Learning Objective	LLOT1560.5
	Question source	Bank 560429
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.5, 41.10
	Comments	Fixed stem typo, and added (to Explanation) a justification for this being an RO-level question, to address NRC Rev. 00 comments.

40 **ID: 1102663** **Points: 1.00**

Unit 1 is operating at 100% power.

The running TECW Pump trips and the standby pump fails to start.

WHICH ONE of the following is an expected plant/system response?

A. Isophase Bus cooler high temperature

B. Main Turbine Lube Oil from cooler high temperature

C. Stator Cooling return high temperature

D. Service Air Compressor aftercooler high temperature

Answer: D

Answer Explanation

The Service Air Compressors are cooled by TECW. The Isophase Bus uses Service Water for cooling, as do Main Turbine Lube Oil and the Stator Cooling Water heat exchangers.

'D' is correct: Service Air Compressor aftercooler high temperature. Correct the reasons described above.

'A' is wrong: Isophase Bus cooler high temperature. Plausible to the examinee who incorrectly concludes that systems/components associated with the Main Turbine/Generator (in the Turbine Enclosure) must use TECW as their source of cooling.

'B' is wrong: Main Turbine Lube Oil from cooler high temperature. Plausible for the same reason as for choice 'A'.

'C' is wrong: Stator Cooling return high temperature. Plausible for the same reason as for choice 'A'.

Question 40 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	1102663
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LLOT0014.3
Topic:	Loss of TECW Pumps - Predict Plant/System Response
Num Field 1:	3.5
Num Field 2:	3.6
Text Field:	295018 AK1.01
Comments:	Level RO Tier 1

	Group	1
	KA # and Rating	295018 AK1.01 (3.5/3.6)
	KA Statement	295018 Partial or Total Loss of CCW Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: AK1.01 Effects on component/system operations
	References	LLOT0014 lesson plan
	Examinee References	None
	Learning Objective	LLOT0014.3
	Question source	Bank 561330
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.4
	Comments	None

41 **ID: 1102684** **Points: 1.00**

Both Units are operating at 100% power.

WHICH ONE of the following will **directly** produce a FULL SCRAM on one of the Units?

A. Unit 1 - Fast Closure of TCVs 1 and 2

B. Unit 1 - Fast Closure of TCV 1 with Closure of TSV 2

C. Unit 2 - Closure of TSVs 2 and 3

D. Unit 2 - Fast Closure of TCV 3 with Closure of TSV 4

Answer: A

Answer Explanation

Refer to the UNIT 1 RPS Elementary Drawings C71-1020-E-006, 007, and 009. Refer to the UNIT 2 RPS Elementary Drawings C71-1020-E-020, Sheets 6, 7, and 8. NOTE - These drawings use the abbreviation "MSV" (Main Stop Valve) as synonomous with TSV.

From these drawings, we can verify the following regarding Unit 1:

- Coincident closure of MSV 3 and 4 trips RPS Channel A1
- Coincident closure of MSV 1 and 2 trips RPS Channel A2
- Coincident closure of MSV 1 and 3 trips RPS Channel B1
- Coincident closure of MSV 2 and 4 trips RPS Channel B2
- Fast closure of TCV 1 trips RPS Channel B1
- Fast closure of TCV 2 trips RPS Channel A1
- Fast closure of TCV 3 trips RPS Channel B2
- Fast closure of TCV 4 trips RPS Channel A2

From this, we discover the following regarding Unit 1:

- No two MSVs are sufficient to satisfy the RPS one-out-of-two-twice logic for a Full Scram; it takes a minimum of three MSVs
- The "right" combination of two TCVs (for example: TCVs 1 and 2) will satisfy the RPS Full Scram logic

From these drawings, we can verify the following regarding Unit 2:

- The MSV logic is identical to that for Unit 1
- Fast closure of TCV 1 trips Channel B1
- Fast closure of TCV 2 trips Channel B2
- Fast closure of TCV 3 trips Channel A1
- Fast closure of TCV 4 trips Channel A2

From this, we discover the following regarding Unit 2:

- No two MSVs are sufficient to satisfy the RPS one-out-of-two-twice logic for a Full Scram; it takes a minimum of three MSVs
- The "right" combination of two TCVs (for example: TCVs 1 and 3) will satisfy the RPS Full Scram logic

'A' is correct: Unit 1 - Fast Closure TCVs 1 and 2. Correct for the reasons described above.

'B' is wrong: Unit 1 - Fast Closure of TCV 1 with Closure of TSV 2. The single TCV can trip only a single RPS Channel; the single TSV can't trip an RPS Channel. Therefore, the combination suggested by this answer choice produces only a half-scram.

'C' is wrong: Unit 2 - Closure of TSVs 2 and 3. Where the "right" combination of TSVs (for example: TSV 3 and 4) can produce a half-scram, no two TSVs (by themselves) can produce a full scram.

'D' is wrong: Unit 2 - Fast Closure of TCV 3 with Closure of TSV 4. The single TCV can trip only a single RPS Channel; the single TSV can't trip an RPS Channel. Therefore, the combination suggested by this answer choice produces only a half-scram.

All of the distracters are plausible because each either suggests that there is "right" combination of 2 TSVs which produce a full scram, or suggests that there is a "right" TSV/TCV combination which can produce a full scram. The turbine trip logic, historically, is the least understood (and so most often confused) of the logics that input to RPS.

Question 41 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	2.50	
System ID:	1102684	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	LLOT0071.IL4	
Topic:	Recognize Turbine Trip logic that produces an RPS Full Scram	
Num Field 1:	3.6	
Num Field 2:	3.7	
Text Field:	295006 AK2.04	
Comments:	Level	RO
	Tier	1
	Group	1
	KA # and Rating	295006 AK2.04 (3.6/3.7)
	KA Statement	295006 SCRAM Knowledge of the interrelations between SCRAM and the following: AK2.04 Turbine trip logic: Plant-Specific
	References	C71-1020-E-006, Rev.25 C71-1020-E-007, Rev.28 C71-1020-E-009, Rev.11 C71-1020-E-020, Sheet 6, Rev.10 C71-1020-E-020, Sheet 7, Rev.11 C71-1020-E-020, Sheet 9, Rev.5
	Examinee References	None
	Learning Objective	LLOT0071.IL4

	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.7
	Comments	None

42 **ID: 1102685** **Points: 1.00**

Unit 2 plant conditions:

- OPCON 3
- 2A RHR is in Shutdown Cooling
- HV-51-2F003A ('A' RHR HX Outlet) is closed
- HIC-51-203A ('A' RHR HX Outlet Bypass) is 30% open
- Reactor water temperature is 275°F, steady

A controller malfunction causes HIC-51-203A to fully close.

WHICH ONE of the following describes the effect on actual reactor water temperature and temperature indicated at TE-51-2N004A ('A' RHR HX Inlet Temp), 10 minutes later?

	Actual Reactor Water Temperature	TE-51-2N004A Indication
A.	Rises	Rises
B.	Rises	Lowers
C.	Lowers	Rises
D.	Lowers	Lowers

Answer: B

Answer Explanation

The F003A (HX Outlet) is already closed; therefore, when the 203A fails closed, all flow through the RHR HX stops (i.e., all flow goes through the HX Bypass, F048A). Thus, with less cooling the actual reactor water temperature will RISE. However, with flow through the HX isolated, temperature in the HX piping will LOWER.

'B' is correct: Rises; Lowers. Correct for the reasons described above.

'A' is wrong: Rises; Rises. The difficult concept (least understood and most counter-intuitive) is that the isolation of flow through the HX results in that inlet temperature lowering. Plausible for that reason.

'C' is wrong: Lowers; Rises. Plausible to the weaker examinee who doesn't understood the flowpaths that result from the failing-closed on the HX inlet valve.

'D' is wrong: Lowers; Lowers. Plausible for the same reason as that for choice 'C'.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 42 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	2.50
System ID:	1102685
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LLOT0051
Topic:	Predict Impact of RHR HX Bypass Valve failing closed while in SDC
Num Field 1:	3.7
Num Field 2:	3.7
Text Field:	205000 A4.07
Comments:	Level RO Tier 2 Group 1 KA # and Rating 205000 A4.07 (3.7/3.7) KA Statement 205000 Shutdown Cooling Ability to manually operate and/or monitor in the control room: A4.07 Reactor temperatures References LLOT0051 lesson plan Examinee References None Learning Objective LLOT0051.14 Question source Bank 562332 Question history None Cognitive level Higher 10 CFR 55 41.7 Comments None

43 **ID: 1102687** **Points: 1.00**

Unit 1 is operating at 100% power.

A logic failure causes an inadvertent HPCI initiation, with the following:

- HPCI injects into the RPV
- White SEAL-IN light is LIT at 10C647; the light will not reset

WHICH ONE of the following:

(1) identifies whether reactor power will reach an APRM Upscale scram setpoint (Yes/No) if operators take no action to terminate the HPCI injection, and
(2) describes the operator action to terminate the HPCI injection, per S55.2.A (HPCI Shutdown from Automatic or Manual Initiation)?

A. (1) No
(2) Depress and hold the Turbine Trip pushbutton, close the HPCI Steam Supply Valve, then release the Turbine Trip pushbutton

B. (1) Yes
(2) Depress and release the HPCI Manual Isolation pushbutton

C. (1) No
(2) Depress and release the HPCI Manual Isolation pushbutton

D. (1) Yes
(2) Depress and hold the Turbine Trip pushbutton, close the HPCI Steam Supply Valve, then release the Turbine Trip pushbutton

Answer: C

Answer Explanation

The LGS simulator model (in fidelity with the actual plant response) shows that reactor power will peak at approximately 107%, well below the two-loop APRM STP Flow-Biased scram setpoint of [0.65 (W) + 61.7%] (clamped at 116.6%), as well as below the Neutron Flux setpoint, clamped at 118.3%. [Refer to alarm response card ARC-MCR-108, B3 to validates these setpoints.]

Per S55.2.A, Steps 4.2 through 4.6, if the SEAL-IN light won't reset (i.e., the initiation signal is still present), the operator must manually isolate HPCI (per Step 4.5.1) in order to shut down the HPCI turbine and stop injection. Conversely, if the SEAL-IN light will reset, the operator is directed to manually trip the turbine (with the Turbine Trip pushbutton), close the HPCI Steam Inlet (1F001), then release the Turbine Trip pushbutton.

'C' is correct: (1) No; (2) Depress and release the HPCI Manual Isolation pushbutton. Correct for the reasons described above.

'A' is wrong: (1) No; (2) Depress and hold the Turbine Trip pushbutton, close the HPCI Steam Supply Valve, then release the Turbine Trip pushbutton. Part (2) is plausible to the examine who either fails to recognize the meaning of the un-resettable SEAL-IN light (i.e., the initiation signal is sealed-in), or who fails to recognize the impact of that sealed-in initiation signal (i.e., that the HPCI turbine will start up, again, after the Trip pushbutton is released).

'B' is wrong: (1) Yes; (2) Depress and release the HPCI Manual Isolation pushbutton. Part (1) is plausible to the examine who believes that the (HPCI) cold water injection is severe enough so as to cause a scram.

'D' is wrong: (1) Yes; (2) Depress and hold the Turbine Trip pushbutton, close the HPCI Steam Supply Valve, then release the Turbine Trip pushbutton. Plausible for the same reasons as that for choices 'A' and 'B'.

Question 43 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	2.50	
System ID:	1102687	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	LGSOPS1540	
Topic:	Inadvertent HPCI Startup - Diagnose, Predict Power Resoonse, and Recall Operator Action	
Num Field 1:	3.9	
Num Field 2:	4.3	
Text Field:	206000 A2.17	
Comments:	Level	RO
	Tier	2
	Group	1
	KA # and Rating	206000 A2.17 (3.9/4.3)
	KA Statement	206000 HPCI Ability to (a) predict the impacts of the following on the HIGH PRESSURE COOLANT INJECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.17 †HPCI inadvertent initiation: BWR-2,3,4
	References	OT-104, Rev.51 S55.2.A, Rev.15
	Examinee References	None
	Learning Objective	LGSOPS1540, no specified objective
	Question source	New
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.5, 41.10
	Comments	None

44 **ID: 1102692** **Points: 1.00**

Unit 1 is operating at 100% power when the following occurs:

- A ground occurs on 1PP02 125V DC Power Distribution Panel 1BD108
- The ground causes the trip of breaker 1PP02-02

WHICH ONE of the following identifies:

(1) the Unit 1 loads that have de-energized, and
(2) the required operator action?

A. (1) Some Unit 1 MCR annunciator panels
(2) Plant operation may continue but operators must closely monitor systems associated with de-energized annunciators

B. (1) Some Unit 1 MCR annunciator panels
(2) Commence a GP-3 normal plant shutdown while closely monitoring systems associated with de-energized annunciators

C. (1) DEHC Primary Overspeed Trip System (DTOPS)
(2) Verify main turbine trip, reactor scram, and enter T-101 (RPV Control)

D. (1) DEHC Primary Overspeed Trip System (DTOPS)
(2) Verify DEHC Backup Overspeed Trip System (TCS) is in control; if not, trip the main turbine, verify reactor scram, and enter T-101 (RPV Control)

Answer: A

Answer Explanation

Refer to S95.9.G U/1 (125/250 VDC BALANCE OF PLANT STATION BATTERY GROUND INVESTIGATION), where Attachment 2, page 3 of 19, shows that some Unit 1 MCR annunciator panels (specifically, panels 003, 107, 108, 109, 111, and 112) are powered from breaker 1PP02-02. Refer to ON-122 (Loss of Main Control Room Annunciators), where step 2.2 directs operators to closely monitor the affected systems; however, there is no requirement for shutting down the plant.

Refer to DEHC lesson plan LGSOPS0031B, pages 18 and 19, which show that DEHC logic has two power supplies: 120V AC RPS distribution panels 1AY160 (the "Primary" power supply) and 1BY160 (the "Backup" power supply). Refer to procedure E-1AY160 (Loss of 1A RPS UPS Power), "Confirming Indication" 1.23, which shows that the MAJOR TROUBLE TURBINE CONTROL annunciator (ARC-MCR-105, B2) may alarm on loss of 1AY160. Refer to the "CAUSES" section of alarm response card ARC-MCR-105, B2, which shows the various potential inputs to this alarm. Refer to alarm response card ARC-MCR-105, D1 (SPEED SIGNALS LOSS OF FEEDBACK TRIP), which shows that a main turbine trip will automatic result from loss of any 2 of the 3 active speed probes in the Unit 1 DEHC logic system (those probes are used for overspeed trip protection).

'A' is correct: (1) Some Unit 1 MCR annunciator panels; (2) Plant operation may continue but operators must closely monitor systems associated with de-energized annunciators. Correct for the reasons described above.

'B' is wrong: (1) Some Unit 1 MCR annunciator panels; (2) Commence a GP-3 normal plant shutdown while closely monitoring systems associated with de-energized annunciators. Plausible because ON-122 is not an often-trained on abnormal operating procedure. Inexperienced operator candidates could easily believe that, in the name of "conservative decision-making", a normal plant shutdown in prudently called-for.

'C' is wrong: (1) DEHC Primary Overspeed Trip System (DTOPS); (2) Verify main turbine trip, reactor scram, and enter T-101 (RPV Control). Part (1) of this choice (as well as for choice 'D') is especially plausible to the examinee who recalls that the "old" Unit 1 EHC logic system (i.e., before the DEHC modification installation) used BOP 125V DC (as opposed to 120V AC) for its logic power. Given that this examinee concludes that the main turbine should have tripped, then Part (2), consequently, becomes the logical required operator action.

'D' is wrong: (1) DEHC Primary Overspeed Trip System (DTOPS); (2) Verify DEHC Backup Overspeed Trip System (TCS) is in control; if not, trip the main turbine, verify reactor scram, and enter T-101 (RPV Control).
Part (2) is plausible to the examinee who, even though he/she incorrectly concludes that the DEHC logic is still DC-powered, that examinee correctly recalls that there is in fact a "Backup" overspeed trip system (the "TCS"); the examinee therefore concludes that it would not make sense to have a backup if we were so quick to trip the main turbine as a result of losing only the "Primary" protection system.

Question 44 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	2.50	
System ID:	1102692	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:		
Topic:	DC Ground - Predict system impact and required action	
Num Field 1:	2.8	
Num Field 2:	3.2	
Text Field:	263000 A2.01	
Comments:	Level	RO
	Tier	2
	Group	1
	KA # and Rating	263000 A2.01 (2.8/3.2)
	KA Statement	263000 DC Electrical Distribution Ability to (a) predict the impacts of the following on the D.C. ELECTRICAL DISTRIBUTION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.01 Grounds

	References	ARC-MCR-226, E1, Rev.3 ON-122, Rev.18 ARC-MCR-105, D1, Rev.1 S95.9.G U/1, Rev.10 E-1AY160, Rev.28 ARC-MCR-105, B2, Rev.3
	Examinee References	None
	Learning Objective	LGSOPS0095, no specified objective
	Question source	New
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.5, 41.10
	Comments	Major revision to address NRC Rev.00 comments. Added additional references to support the revision.

45 **ID: 1104186** **Points: 1.00**

Unit 1 is operating at 100% power, steady.

The 'B' CRD Flow Control Valve (F002B) is in service when the following occurs:

- Instrument Air tubing breaks off at its connection to the F002B valve positioner

WHICH ONE of the following describes a resulting CRD related indication at 10C603?

A. DRIVE WATER D/P reads fully upscale

B. CHARGING WATER PRESSURE reads lower than normal

C. F002B Green light ON, Red light OFF

D. Flow Controller OUTPUT signal reads 100%

Answer: D

Answer Explanation

Refer to P&ID M-0046, Sheet 001, coordinates C-D/2, which shows that the CRD FCVs (F002A/B) fail closed (F.C.) on loss of air.

'D' is correct: Flow Controller OUTPUT signal reads 100%. When the FCV fails closed, the Flow Controller (FC 1R600) "sees" the dramatic reduction in flow through its Flow Element (FE 1N003) and responds by attempting to restore system flow to its normal Flow Controller setpoint of approximately 60 gpm. As such, its OUTPUT signal goes from a pre-failure value of about 30% to a fully-upscale value of 100%.

'A' is wrong: DRIVE WATER D/P reads fully upscale. Normal Drive Water D/P reads about 250 psid, established by the combination of the in-service FCV position and the throttled position of the Pressure Control Valve, F003. When F002B fails closed, the Drive Water d/p goes to essentially 0 psid. Plausible to the examinee who either believes that F002B is a fail-open valve, or who does not comprehend the cause-effect relationship between F002B position and Drive Water d/p.

'B' is wrong: CHARGING WATER PRESSURE reads lower than normal. Normally, there is no flow through the Charging Water Header and so the Charging Water Pressure indicator reads essentially the discharge pressure (minus some amount of head loss) of the running CRD pump (about 1400 psig). When F002B fails closed, the reduced flow through the Cooling Water Header results in a slight (approximately 40 psig) rise in the "backpressure" felt by the Charging Water Header pressure transmitter. Plausible to the examinee who lacks a solid comprehension of these system dynamics, or who simply believes that F002B fails open.

'C' is wrong: F002B Green light ON, Red light OFF. The F002 valves have a mechanical stop near their closed seats, preventing the valve disk from ever going fully closed (thereby esnuring a minimal amount of cooling water flow is always going to the CRDMs. As such, the FCV always has dual-position indication (Green light ON, Red light ON). Plausible to the examinee who recall that F002B fails closed, but forgets about the mechanical stop near its closed seat.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 45 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	2.50
System ID:	1104186
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LLOT0046.9
Topic:	Loss of Instrument Air to In-service CRD FCV - Predict resulting CRD indications
Num Field 1:	3.3
Num Field 2:	3.4
Text Field:	300000 K3.02
Comments:	Level RO Tier 2 Group 1 KA # and Rating 300000 K3.02 (3.3/3.4) KA Statement 300000 Instrument Air Knowledge of the effect that a loss or malfunction of the (INSTRUMENT AIR SYSTEM) will have on the following: K3.02 Systems having pneumatic valves and controls References P&ID M-0046, Sheet 001, Rev.51 Examinee References None Learning Objective LLOT0046.9 Question source New Question history None Cognitive level Higher 10 CFR 55 41.7 Comments None

46 **ID: 1102745** **Points: 1.00**

WHICH ONE of the following identifies the circuit breaker that supplies power to the solenoid for the Unit 1 ARI Valve SV-1F163A?

A. 1PP01-09

B. 1PPA1-09

C. 10Y101-32

D. 1AY160-32

Answer: B

Answer Explanation

Each ARI valve has a single DC-powered solenoid (refer to P&ID M-0047, sheet 1, coordinate H-5 to validate for two of these valves (SV-1F163A/(B)). The solenoid receives its power from its respective Division (1 or 2) of the Redundant Reactivty Control System (RRCS), which receives its power from the respective Division (1 or 2) 125 VDC bus. In the context of this question, refer to RRCS valve/breaker alignment procedure S36.1.A (COL), page 3, Step 1, which shows that 125 VDC breaker 1PPA1-09 has the "Apparatus Description" of: "RRCS PNL BREAKER 10C634 29-A10209." Refer to RRCS Elementary Drawing C22-1050-E-102, sheet 3, which shows that the Channel 'A' (i.e., Div 1) ARI Valve solenoids are powered from the 124 VDC Bus 'A' panel 1AD102, circuit #9. This description is synonomous with the 1PPA1-09 label for this breaker.

'B' is correct: 1PPA1-09. Correct for the reasons described above.

'A' is wrong: 1PP01-09. This choice refers to breaker #9 of 125 VDC distribution panel 1PP01, which is a Non-Safeguards (non-Divisional) bus, not a Division bus. Refer to alarm response card ARC-MCR-125, A3 to validate. Because this Non-Safeguards DC panel does in fact supply some of the power requirements for RRCS (e.g., as shown for Step 8 of the S36.1.A (COL) procedure cited earlier), this choice is plausible to the examinee who mistakenly believes that a non-Divisional bus might in fact supply power to Divisional (Safeguards) related components, such as the ARI Valve solenoids. It may well be that the examinee does not even recognize that those solenoids are in fact "safety-related" components.

'C' is wrong: 10Y101-32. This is a 120 VAC distribution panel, not a DC one. Plausible to the examinee who fails to recall that the ARI Valve solenoids are DC-powered. This panel/breaker does in fact supply some of the power to RRCS, as shown at Step 4 of the same procedure cited earlier.

'D' is wrong: 1AY160-32. This is the 120 VAC RPS System 'A' distribution panel, not a DC panel. Plausible to the examinee who fails to recall that the ARI Valve solenoids are DC-powered, and so mistakenly concludes that it would make sense that the solenoids are powered from the same source as the RPS scram solenoid pilot valves.

Question 46 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	2.50	
System ID:	1102745	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	LLOT0046	
Topic:	Recognize power supply to the ARI Valve solenoids	
Num Field 1:	4.5	
Num Field 2:	4.5	
Text Field:	201001 K2.05	
Comments:	Level	RO
	Tier	2
	Group	2
	KA # and Rating	201001 K2.05 (4.5/4.5)
	KA Statement	201001 CRD Hydraulic Knowledge of electrical power supplies to the following: K2.05 Alternate rod insertion valve solenoids: Plant-Specific
	References	P&ID M-0047, sheet 1, Rev.47 C22-1050-E-102, sheet 3, Rev.6 1S36.1.A (COL), Rev.4 ARC-MCR-125, A3, Rev.0
	Examinee References	None
	Learning Objective	LLOT0046, no specified objective
	Question source	New
	Question history	None
	Cognitive level	Higher*
	10 CFR 55	41.7
	Comments	*Justification for Higher Cognitive categorization: The answer choices require the examinee to recognize that the 1PPA1 label designates a Divisional (Safeguards) bus, whereas the 1PP01 label designates a non-Safeguards bus. The question also requires the examinee to recognize that the solenoids are in fact "Safety-related" components, thereby warranting Safeguards electrical power, and that they are DC-powered components.

47 **ID: 1102751** **Points: 1.00**

Unit 2 is operating at 100% power when the following occurs:

- One cell fails (and is bypassed) on the 2A ASD

No operator action is taken.

WHICH ONE of the following:

(1) describes how **reactor power** responds, and
(2) identifies whether a 2A ASD SPEED HOLD alarm is sealed-in (Yes/No)?

A. (1) Lowers approximately 2-3%
(2) No

B. (1) Remains unchanged
(2) Yes

C. (1) Lowers approximately 2-3%
(2) Yes

D. (1) Remains unchanged
(2) No

Answer: C

Answer Explanation

Refer to S43.1.F, Attachment 2 FRR Alarm List, FRR # 2.1.1 (Cell Fault), which shows that a single cell fault allows the ASD to continue operating but with a Speed Hold having been automatically applied (locking up the ASD at its resulting speed)...refer to alarm response card ARC-MCR-211, C3, for the alarm that will accompany the Speed Hold. What's not mentioned on this Alarm List table is that the cell failure causes a speed drop of about 100-200 rpm. This speed drops cause Total Core Flow to drop enough so as to result in a reactor power drop of about 2-3%.

NOTE - The following is an excerpt from the "Cause & Effects" description provided in the LGS Simulator model for Malfunction # MRR203, which describes the ASD speed drop that occurs when an ASD cell fault occurs. The entire Cause & Effects description for this Malfunction is included in the folder titled "Added Exam References" that is part of this NRC Exam Submittal...

A cell fault occurs due to the activation of one of its internal protection mechanisms. The VFD will bypass the cell and keep operating until there are not enough available cells to support the load, in such a case a System Fault will occur. (Trip: The total number of active cells in the drive / 3 where at least one cell must be available in each phase is less then Menu ID 2540.) (Ref: FRR page 55). ***A power cell trip will result in an interruption of the ASD output current to the motor of up to 250 msec. The pump and motor coasts during this interruption. Actual data from a cell bypass with the reactor at 100% power shows Recirc Pump speed dropped by 8% (134 rpm) over 2 seconds and then raised by 2.4% (40 rpm) over 2 seconds.***

'C' is correct: (1) Lowers approximately 2-3%; (2) Yes. Correct for the reasons described above.

'A' is wrong: (1) Lowers approximately 2-3%; (2) No. Part (2) is plausible to the examinee who does recognize that the ASD remains operating (i.e., doesn't trip) but who fails to recall the Speed Hold that results from the cell failure.

'B' is wrong: (1) Remains unchanged; (2) Yes. Part (1) is plausible to the examinee who does recognize that the ASD remains operating but who fails to recall the speed drop (and, therefore, the reactor power drop) that occurs from the cell failure.

'D' is wrong: (1) Remains unchanged; (2) No. Plausible for reasons similar to that for choices 'A' and 'B'.

Question 47 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	1102751
User-Defined ID:	REV 01, 12/15/14
Cross Reference Number:	
Topic:	Predict reactor power response to ASD single-cell failure
Num Field 1:	3.4
Num Field 2:	3.4
Text Field:	202002 A3.02
Comments:	Level: RO Tier: 2 Group: 2 KA # and Rating: 202002 A3.02 (3.4/3.4) KA Statement: 202002 Recirc Flow Control Ability to monitor automatic operations of the RECIRCULATION FLOW CONTROL SYSTEM including: A3.02 Lights and alarms References: S43.1.F, Rev.1 ARC-MCR-211, C3, Rev.2 Examinee References: None Learning Objective: LGSOPS0043B.2C Question source: New Question history: None Cognitive level: Lower 10 CFR 55: 41.7 Comments: None

48 **ID: 1102752** **Points: 1.00**

A localized Instrument Air line break results in the loss of air to the Unit 2 Fuel Pool Cooling and Cleanup System.

WHICH ONE of the following is an expected plant/system response?

A. Spent Fuel Pool water clarity degrades

B. Spent Fuel Pool water temperature rises

C. Spent Fuel Pool level lowers

D. Skimmer Surge Tank level lowers

Answer: A

Answer Explanation

The Skimmer Surge Tank Makeup Valve (HV-53-207, an AOV) fails open on a loss of air; therefore, Demin Water is continuously supplied to the tank, causing its level to rise, not lower (refer to P&ID M-0053, sheet 4, coord. H-3). The loss of air has no impact on Spent Fuel Pool level itself. The FPCC Filter-Demin Valves (AOVs; FV-C 203, for example) fail closed on a loss of air; thus, spent fuel pool water filtration stops, resulting in a degradation of the water clarity (refer to P&ID M-0054, sheet 2, coord. F-6). Because the Filter-Demin Valves fail so as to stop flow through the Filer-Demins, the Filter-Demin Bypass Valve (HV-54-205, an AOV) is designed to fail open, to ensure continued system operation to maintain spent fuel pool water temperature unchanged (refer to P&ID M-0053, sheet 4, coord. E-2).

'A' is correct: Spent Fuel Pool water clarity degrades. Correct for the reasons described above.

'B' is wrong: Spent Fuel Pool water temperature rises. Plausible to the examinee who either does not recall the failure mode for the F-D Bypass Valve, or who simply doesn't recall that the valve exists.

'C' is wrong: Spent Fuel Pool level lowers. Plausible to the examinee who mistakenly believes that the skimmer surge tank makeup valve fails closed; as such, the surge tank level lowers and so too does the spent fuel pool level.

'D' is wrong: Skimmer Surge Tank level lowers. Plausible to the examinee who fails to recall the failure mode for the skimmer surge tank makeup valve ,as described above.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 48 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	1102752
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LGSOPS0053
Topic:	Predict Loss of Inst Air impact on FPCC System
Num Field 1:	2.6
Num Field 2:	2.8
Text Field:	233000 K3.03
Comments:	Level RO Tier 2 Group 2 KA # and Rating 233000 K3.03 (2.6/2.8) KA Statement 233000 Fuel Pool Cooling/Cleanup Knowledge of the effect that a loss or malfunction of the FUEL POOL COOLING AND CLEAN-UP will have on following: K3.03 Fuel pool water clarity References P&ID M-0053, sheet 4, Rev.17 P&ID M-0054, sheet 2, Rev.7 Examinee References None Learning Objective LGSOPS0053, no specified objective Question source New Question history None Cognitive level Higher 10 CFR 55 41.7 Comments None

49 **ID: 1116786** **Points: 1.00**

Unit 1 is in OPCON 1.

Operators are performing ST-6-001-760-1 (Main Stop Valve and CIV Exercise Test).

As directed by the ST, the PRO:

- Selects the MSV #1 TEST window on the 10C653 DEHC HMI, then
- Presses the START button on that window

Consider the following plant parameters:

1. RPV water level
2. Main Condenser vacuum
3. Main Generator output (MWe)

WHICH ONE of the following identifies **ALL** of the plant parameters (from the above list) that **LOWER** while MSV-1 is stroking in the **CLOSED** direction?

A. 1

B. 1 and 2

C. 2 and 3

D. 1, 2, and 3

Answer: D

Answer Explanation

Refer to ST-6-001-760-1, the following sections:

- Prerequisite 2.5 requires reactor power to be at or below 92% prior to performing the test
- Step 3.6 reminds us that about a 4% spike (rise) in reactor power is expected when an MSV is stroking closed

A consideration for basic BWR Generic Fundmentals reminds of the following:

- The spike up in reactor power is due to the rise in RPV pressure that results from closing off one of the 4 main steam lines going to the main turbine.
- The RPV pressure rise will collapse some core "voids", causing an **indicated RPV level drop** as some of the water sweeps from from the downcomer annulus back into the core.

Not so evident from an understanding of Generic Fundamentals is the impact on Main Generator real power (Megawatts) and Main condenser vacuum. The following plant dynamics occur when the MSV strokes closed:

- The rise in RPV pressure translates to a corresponding rise in main steam line pressure averaging manifold (PAM) pressure, which causes the EHC Pressure Regulator to throttle open on the Turbine Control Valves (TCVs) in an effort to restore/maintain RPV pressure as required by the EHC "Pressure Set". However, the TCVs can't open enough to maintain PAM pressure, by themselves; thus, the EHC Bypass Control Unit causes the Turbine Bypass Valves (TBVs) to open enough to help the TCVs restore pressure to Pressure Set. Problem is...the now-open TBVs rob some amount of main steam away from the HP turbine; the result is a **drop in Main Generator load (MWe).**
- Because the TBVs are now dumping some steam into the **main condenser, its vacuum lowers**.

'D' is correct: 1, 2, and 3. Correct for the reasons described above.

'A' is wrong: 1.

'B' is wrong: 1 and 2.

'C' is wrong: 2 and 3.

Distracters are all plausible for the reasons suggested in the comprehensive explanation of the plant/system dynamics, above. At the point of sitting for an NRC Exam, history has shown that ILT RO/SRO Candidates are far removed from their former solid-grounding in the plant behaviors they learned in their GFE Phase of training.

Question 49 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	3.50	
System ID:	1116786	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	NONE	
Topic:	Predict plant parameter changes as a result of stroke-testing an MSV	
Num Field 1:	3.7	
Num Field 2:	3.7	
Text Field:	239001 A1.07	
Comments:	Level	RO
	Tier	2
	Group	2
	KA # and Rating	239001 A1.07 (3.7/3.7)
	KA Statement	239001 Main and Reheat Steam Ability to predict and/or monitor changes in parameters associated with operating the MAIN AND REHEAT STEAM SYSTEM controls including: A1.07 Reactor water level
	References	ST-6-001-760-1, Rev.42
	Examinee References	None

	Learning Objective	No specified objective
	Question source	New
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.5
	Comments	None

50 **ID: 1102765** **Points: 1.00**

Both Units are in OPCON 1 when the following occurs:

- All 4 radiation monitor channels sense a valid high radiation condition on the Control Room HVAC intakes
- No Control Room HVAC automatic isolation occurs

CRS directs the PRO to manually initiate a Control Room radiation isolation, per S78.8.A (Manual Initiation of Control Room Radiation or Chlorine/Toxic Chemical Isolation).

Consider the following list of possible action steps to accomplish this task:

1. Place both Control Room Isolation Valve Reset Keylock Switches to RESET
2. Place both Control Room Isolation Valve Trip Switches to RAD
3. Place both Control Room Isolation Valve Reset Keylock Switches to AUTO
4. Depress and release the pushbutton portion of both Trip Switches

WHICH ONE of the following identifies the required actions in the **specific sequence required** by S78.8.A?

A. Perform Steps 1,2,3,4 in that order for BOTH Subsystems (A and B)

B. Perform Steps 1,2,3,4 in that order for just ONE Subsystem (A or B)

C. Perform Steps 2,1,4 in that order for BOTH Subsystems (A and B)

D. Perform Steps 2,1,4 in that order for just ONE Subsystem (A or B)

Answer: A

Answer Explanation

Refer to S78.8.A, the following sections:

- NOTE #2 (on page 4) directs operators to manually initiate BOTH subsystems (A and B)
- Section 4.2 manually initiates a radiation isolation of the 'A' subsystem, as follows:

(Step 4.2.2) - Place both Control Room Isolation Valve Reset Keylock Switches to RESET
(Step 4.2.3) - Place both Control Room Isolation Valve Trip Switches to RAD
(Step 4.2.4) - Place both Control Room Isolation Valve Reset Keylock Switches to AUTO
(Step 4.2.5) - Depress and release the pushbutton portion of both Trip Switches

'A' is correct: Perform Steps 1,2,3,4 in that order for BOTH Subsystems (A and B). This answer choice includes the exact sequence of actions described above, and for BOTH subsystems, as required by S78.8.A.

'B' is wrong: Perform Steps 1,2,3,4 in that order for just ONE Subsystem (A or B). This choices includes the exact sequence of actions described above; however, it suggests that only one Subsystem needs to be manually isolated. Plausible, especially, to inexperienced operator/operator candidates who have had minimal training time in the simulator with respect to performing this task. What makes it very distracting is that the examinee does recall that, if the automatic isolation had occurred (as it should have), only one CREFAS Fan will have started (i.e., in the normal Standby lineup for CREFAS, one Fan is in AUTO...meaning that it will auto-start in response to the high radiation signal...while the other Fan is in Standby...meaning that it will start (after a time delay) only if the AUTO Fan fails to start or otherwise produces a low-flow condition).

'C' is wrong: Perform Steps 2,1,4 in that order for BOTH Subsystems (A and B). This choice suggests that the operator "Place both Control Room Isolation Valve Trip Switches to RAD" before "Placing both Control Room Isolation Valve Reset Keylock Switches to RESET...". The normal position for the Control Room Isolation Valve Reset Keylock Switches is AUTO. If the operator fails to place those switches in RESET, before rotating the Trip Switch collars to the RAD position, the Subsystem will go into a Chlorine (rather than a Radiation) isolation lineup. This choice is plausible for reasons similar to that for choice 'B' (lack of familiarity with the task). It's also plausible to the examinee who lacks an understanding of the isolation logic, especially as it relates to the function of the Valve Reset Keylock Switches.

'D' is wrong: Perform Steps 2,1,4 in that order for just ONE Subsystem (A or B). This choice is identical to choice 'C', except that it suggests that only one Subsystem needs to be isolated. Plausible for the same reasons as for choices 'B' and 'C'.

Question 50 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	5	
Difficulty:	2.50	
System ID:	1102765	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LGSOPS0078.IL7	
Topic:	Recall how to manually initiate a control room radiation isolation	
Num Field 1:	3.2	
Num Field 2:	3.2	
Text Field:	290003 A4.01	
Comments:	Level	RO
	Tier	2
	Group	2
	KA # and Rating	290003 A4.01 (3.2/3.2)
	KA Statement	290003 Control Room HVAC Ability to manually operate and/or monitor in the control room: A4.01 Initiate/reset system
	References	S78.8.A, Rev.16
	Examinee References	None
	Learning Objective	LGSOPS0078.IL7
	Question source	New
	Question history	None

	Cognitive level	Lower
	10 CFR 55	41.7, 41.10
	Comments	Significant editorial re-construction of the stem and answer choices to address NRC Rev.00 comments.

51 **ID: 1118326** **Points: 1.00**

Plant conditions:

- Unit 1 is operating at 100% power
- Unit 2 is in OPCON 5 for a refueling outage
- "205" Breaker is blocked for maintenance
- The 3rd offsite source is NOT yet connected
- ALL Fast Transfer Select Switches are in their normal alignment

The following occurs:

- A valid 1 UNIT PROTECTION RELAYS ENERGIZED alarm is received (ARC-MCR-125 GEN 1, E1)

No operator action is taken.

WHICH ONE of the following identifies a status of the Unit 1 electrical distribution system?

A. 10-11 Breaker is CLOSED
20-12 Breaker is CLOSED

B. 10-11 Breaker is CLOSED
20-12 Breaker is OPEN

C. 10-12 Breaker is CLOSED
20-11 Breaker is CLOSED

D. 10-12 Breaker is CLOSED
20-11 Breaker is OPEN

Answer: B

Answer Explanation

Examinee is expected to recognize that the valid "1 UNIT PROTECTION RELAYS ENERGIZED" alarm (ARC-MCR-125, E1) amounts to a Unit 1 Main Generator Lockout condition. The alarm response card shows the automatic action designed to occur on a Main Generator lockout, including: Both generator OUTPUT breakers (535 and 635) trip; the Alterrex Exciter FIELD BREAKER trips; the 11 and 12 Unit Aux Buses fast-transfer to their pre-selected Startup Buses.

Normally (with both Units operating at 100% power), the Unit 1 Aux Buses (11 and 12) are powered directly from the Unit 1 Main Generator via the 11 Unit Aux Transformer. The 11 Unit Aux Bus is aligned (by way of a "Fast Transfer Select" switch at panel 10C654) to fast-transfer to the 10 Startup Bus by way of the 10-11 Breaker automatically closing, and the 12 Unit Aux Bus is pre-selected (with its own Fast Transfer Select switch) to fast-transfer to the 20 Startup Bus by way of the 20-12 Breaker automatically closing. Stem conditions indicate that these two switches are still in their normal alignment.

The "205" normally supplies the 20 Startup Bus; however, with that Breaker blocked (cleared & tagged) out of service for maintenance, the 20 Startup is not available (dead bus; no voltage). Thus, on the Unit 1 Main Generator Lockout, the 20-12 Breaker will NOT automatically close (i.e., no Fast Transfer of the 12 Bus to the 20 Startup Bus occurs). In other words, the following breaker alignment results: 10-11 Breaker is CLOSED; 20-12 Breaker is OPEN.

'B' is correct: 10-11 Breaker is CLOSED; 20-12 Breaker is OPEN. Correct for the reasons described above.

'A' is wrong: 10-11 Breaker is CLOSED; 20-12 Breaker is CLOSED. Plausible to the examinee who does not recognize that the 205 Breaker outage has the 20 Startup Bus de-energized and, therefore, the Fast Transfer of the 12 Bus over to the 20 Startup Bus will NOT occur.

'C' is wrong: 10-12 Breaker is CLOSED; 20-11 Breaker is CLOSED. This choice is plausible to the examinee who incorrectly recalls how the Fast Transfer Select Switches are normally aligned; that examine instead believes that the 11 Bus is aligned to transfer to the 20 Startup Bus and the 12 Bus is aligned to transfer to the 10 Startup Bus; they are NOT.

'D' is wrong: 10-12 Breaker is CLOSED; 20-11 Breaker is OPEN. Plausible for reasons similar to that for choice 'C'. In this case, however, the examinee does recognize that the de-energized 20 Startup Bus will prevent the Fast Transfer of the 11 Bus from occurring.

Question 51 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	3.00	
System ID:	1118326	
User-Defined ID:	REV 02, 01/06/15	
Cross Reference Number:	LLOT0035.3	
Topic:	Predict Unit 1 electrical distriibution status after Unit 1 Protection Relays actuation	
Num Field 1:	3.2	
Num Field 2:	3.3	
Text Field:	295005 AA2.08	
Comments:	Level	RO
	Tier	1
	Group	1
	KA # and Rating	295005 AA2.08 (3.2/3.3)
	KA Statement	295005 Main Turbine Generator Trip Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: AA2.08 Electrical distribution status
	References	ARC-MCR-125, E1, Rev.0
	Examinee References	None
	Learning Objective	LLOT0035.3
	Question source	New
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.4

	Comments	Major revision to stem and all answer choices, to address NRC Rev.00 comments. Major revision, again (Rev.02), to resolve concerns identified during Validation.

52 **ID: 1103365** **Points: 1.00**

Unit 1 is in OPCON 3.

RHR 'A' is in Shutdown Cooling.

WHICH ONE of the following will result in an ON-121 (Loss of Shutdown Cooling) entry?

A. Loss of 1AY160

B. Loss of 1AY185

C. RPV level 27.5"

D. RPV pressure 64.5 psig

Answer: A

Answer Explanation

Refer to ON-121, Section 2.1 NOTE. A loss of RPS UPS 1AY160 results in an NSSSS automatic isolation of the RHR Suction Isolation Valves (HV-051-1F008 and 009) and the RHR Return Valves (HV-051-1F015(B)).

Refer to GP-8 (Primary and Secondary Containment Isolation Verification and Reset), Attachment 1, page 1 of 3, and Attachment 2, pages 1 and 2, which show that the RHR SDC Isolation Group IIA has two isolation signals ('A' and 'V'): RPV Level 3 (+12.5"), and RPV High Pressure (75 psig).

'A' is correct: Loss of 1AY160. Correct for the reasons described above.

'B' is wrong: Loss of 1AY185. This is one of the two APRM UPS's (1AY185 and 1BY185), neither of which powers any of the NSSSS isolation logic. Plausible to the examinee who recalls that the NSSSS logic is in fact UPS-powered, but who confuses the APRM UPS's with the RPS UPS's.

'C' is wrong: RPV level 27.5". This is RPV Level 4 (the low level alarm, only). Plausible to the examinee who incorrectly recalls/confuses Level 3 with Level 4.

'D' is wrong: RPV pressure 64.5 psig. This is the pressure at which RCIC will automatically isolate. Plausible to the examinee who confuses the 64.5 psig RCIC islolation setpoint with the SDC isolation setpoint of 75 psig.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 52 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.00	
System ID:	1103365	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LGSOPS0051.14A	
Topic:	Recall RHR SDC Isolations	
Num Field 1:	4.0	
Num Field 2:	4.2	
Text Field:	295021 2.4.11	
Comments:	Level	RO
	Tier	1
	Group	1
	KA # and Rating	295021 2.4.11 (4.0/4.2)
	KA Statement	295021 Loss of Shutdown Cooling 2.4.11 Knowledge of abnormal condition procedures.
	References	ON-121, Rev.29 GP-8 U/1, Rev.16
	Examinee References	None
	Learning Objective	LGSOPS0051.14a
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.7
	Comments	Minor revision (replaced 'B' distractor) to address NRC Rev.00 comments.

53 **ID: 1103385** **Points: 1.00**

Unit 1 plant conditions:

- OPCON 5
- Core Shuffle Part 2 is in progress
- Fuel bundle 43-20 has just been seated in the core
- The main hoist grapple is released and is being raised

Observing the SRMs, the RO determines that an inadvertent criticality has resulted from this fuel bundle movement.

WHICH ONE of the following describes the required action per ON-120 (Fuel Handling Problems)?

A. Notify Health Physics to determine dose rates

B. Re-grapple fuel bundle 43-20 and raise it until it clears the top guide

C. Direct a second licensed operator to verify that the SRMs are not noise-spiking

D. Evacuate the fuel floor and ensure all insertable control rods are inserted

Answer: D

Answer Explanation

Refer to ON-120, Section 2.1. Because the SRM count rate is continuing to rise, Step 2.1.4 applies.

'D' is correct: Evacuate the fuel floor and ensure all insertable control rods are inserted. Correct for the reason described above.

'A' is wrong: Notify Health Physics to determine dose rates. Plausible to the examinee who does not recognize that the SRM count rate has more than doubled and has not stabilized and is increasing, indicating criticality. Condition exists for an evacuation and HP is notified to assist with evacuation.

'B' is wrong: Re-grapple fuel bundle 43-20 and raise it until it clears the top guide. Plausible to the examinee who does not recognize that the SRM count rate has more than doubled and has not stabilized and is increasing, indicating criticality. If the count rate had stabilized then the correct action would be to raise the bundle until it clears the upper guide, however, after grapple has been released there is no direction to re-grapple.

'C' is wrong: Direct a second licensed operator to verify that the SRMs are not noise-spiking. Plausible to the examinee who only recalls the FH-105 (Core Component Movement), Section 3.8.5.2 concern for distinguishing between...an SRM trend (two doublings) that in fact is indicative of an inadvertent criticality...and noise-induced spiking of the SRMs. Uncertain of him/herself, this examinee choses to proceed with caution before ordering an evacuation.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 53 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	2.50
System ID:	1103385
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LLOT0760.10
Topic:	Recall ON-120 actions for signs of inadvertent criticality
Num Field 1:	3.7
Num Field 2:	4.0
Text Field:	295023 AK1.03
Comments:	Level RO Tier 1 Group 1 KA # and Rating 295023 AK1.03 (3.7/4.0) KA Statement 295023 Refueling Accident Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS: AK1.03 Inadvertent criticality References ON-120, Rev.24 Examinee References None Learning Objective LLOT0760.10 Question source Bank 1097324 Question history LGS 2012 NRC Exam Question #10 Cognitive level Lower 10 CFR 55 41.10, 41.11 Comments None

54 **ID: 1103389** **Points: 1.00**

Consider the requirements of OP-LG-103-102-100, Human Performance Continuing Good Practices.

WHICH ONE of the following identifies the computer-based system an operator is directed to use to verify the revision of a Surveillance Procedure (ST)?

A. EDMS

B. PIMS

C. HPI

D. Passport

Answer: B

Answer Explanation

Refer to OP-LGS-103-102-1000, Attachment 11, "Procedure Use" section, 1st bullet..."current revision is to be checked in PIMS (i.e., "Plant Information Management System").

'B' is correct: PIMS. Correct for the reason described above.

'A' is wrong: EDMS. The "Electronic Data Management System" is the central computer-based system from which all procedures (and most other controlled plant documents) are retrieved and printed for use. However, EDMS is not to be used to verify that the revision of the document contained there is the current one.

'C' is wrong: HPI. The "Human Performance Interface" is another gateway/access portal to EDMS, constructed to make easier and faster access to EDMS documents, along with more convenient search methods. Because it is essentially treated the same way as EDMS, it too is not to be used to verify document revisions.

'D' is wrong: Passport. Passport is the central computer database for plant components/equipment, as well as for managing the Corrective Action System. Although it also contains plant controlled document information, in the form of links/cross-references, it is not a reliable source of information for verifying document revisions.

Distracters are all plausible especially to the inexperienced LO Candidate with little hands-on knowledge of procedure usage.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 54 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	2.00
System ID:	1103389
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LGSOPS2005A.7
Topic:	Recall how to verify a procedure revision
Num Field 1:	3.5
Num Field 2:	3.6
Text Field:	2.1.21
Comments:	Level RO Tier 3 Group N/A KA # and Rating 2.1.21 (3.5/3.6) KA Statement 2.1.21 Ability to verify the controlled procedure copy References OP-LG-103-102-1000, Rev.54 Examinee References None Learning Objective LGSOPS2005A.7 Question source New Question history None Cognitive level Lower 10 CFR 55 41.10 Comments None

55 **ID: 1103390** **Points: 1.00**

Consider the following clearance tag attachment methods:

1. Duct tape attaching a DANGER tag to a manual disconnect in the 500 KV switchyard
2. String attaching an SCT to a lifted lead inside a MCR panel
3. Adhesive ("Scotch") tape attaching an INFORMATION tag to an annunciator window inside the Radwaste Control Room

WHICH ONE of the following identifies (from the above list) method(s), if any, that **COMPLY WITH** the requirements of OP-MA-109-101 (Clearance and Tagging)?

A. 1, 2, and 3

B. 2 and 3, only

C. 3, only

D. None

Answer: B

Answer Explanation

Refer to OP-MA-109-101. Answer choices are taken directly from Section 10.2.1...

- 10.2.1. Danger tags or SCT's will be securely fastened to all points of isolation with a nonreusable nylon cable tie with a minimum unlocking strength of 50 pounds for regular size Tags.

  1. When attachment to the point of isolation is not possible, then the tag shall be attached in a highly visible location as close as possible to the equipment to alert all personnel of tagout conditions.

  2. When tag attachment via a nylon cable tie is impractical, then other methods such as tape, string, or other suitable methods of attachment may be used for tag attachment provided the clearance tag will be in a controlled environment.

  EXAMPLES OF ALTERNATE TAG ATTACHMENT DEVICES

  - String used to attach a clearance tag to a lifted lead inside a control room panel.
  - Adhesive tape used to attach a clearance tag to a radwaste control room annunciator window.

'B' is correct: 2 and 3, only. The manual disconnect is entirely a mechanical device (i.e., no doored-cabinet with electrical controls within); thus, the DANGER tag would be subjected to the outside evironment's weather, meaning it is NOT a controlled environment. As such, the DUCT tape is unacceptable.

'A' is wrong: 1, 2, and 3. Plausible to the examinee who doesn't recognize that Cooling Tower manhole is NOT a controlled environment.

'C' is wrong: 3, only. Plausible to the examinee who recognizes that mechanical disconnect is NOT in a controlled environment, but who also dismisses the "string" method as being acceptable, concluding that a simple string performing such an important function as keeping an electrical lead de-termed is unacceptable, as well.

'D' is wrong: None. Plausible for reasons similar to that for choices 'A' and 'C'; but htis examinee also suspects that the use of "Scotch" tape (not known for its strength) also seems unacceptable.

Question 55 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	2	
Difficulty:	2.00	
System ID:	1103390	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	OP-MA-109-101.7	
Topic:	Clearances - Recall requirements for acceptable tag attachment	
Num Field 1:	4.1	
Num Field 2:	4.3	
Text Field:	2.2.13	
Comments:	Level	RO
	Tier	3
	Group	N/A
	KA # and Rating	2.2.13 (4.1/4.3)
	KA Statement	2.2.13 Knowledge of tagging and clearance procedures.
	References	OP-MA-109-101, Rev.20
	Examinee References	None
	Learning Objective	OP-MA-109-101.7
	Question source	Bank 987013
	Question history	None
	Cognitive level	Higher*
	10 CFR 55	41.10
	Comments	*Justification for Higher Cognitive categorization: Examinee must recognize that the stem condition 1st bullet (mechanical discinnect) is NOT a controlled environment; therefore, the DUCT tape is unacceptable. --- Minor revision (changed one stem condition bullet) to address NRC Rev.00 comments.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

56 **ID: 1104787** **Points: 1.00**

Unit 2 is operating at 100% power, with the following:

- TIP operation is in progress per S74.0.A (Operation of the TIP System)
- The 2E TIP detector is currently inserted in the core

The Reactor Enclosure EO reports that there are workers on the TIP Room roof.

WHICH ONE of the following actions is **immediately** required per S74.0.A?

A. Direct Security to enter the area and evacuate workers

B. Direct the EO to enter the area and evacuate workers

C. Withdraw the 2E TIP into its shield

D. Stop operation of TIP mechanisms

Answer: D

Answer Explanation

S74.0.A, Section 4.2 WARNING states: "TIP detectors shall **not** be moved from their chamber shields until Health Physics has taken appropriate actions for TIP system operation." This WARNING is considered to extend to a TIP detector that is in-core, as well. The intent is that so long as the TIP detector is somehow shielded (be it in-shield, or in-core), dose rates are minimized by preventing the movement of highly irradiated TIP detectors through normally accessible areas. Step 4.3 requires the following PA announcement: "Unit 1 (2) TIPS will be (are) in operation. Please stay clear of the TIP room, the TIP room roof, and affected areas."

'D' is correct: Stop operation of TIP mechanisms. Correct for the reasons described above.

'A' is wrong: Direct Security to enter the area and evacuate workers. No such immediate requirement exists. Plausible to the examinee who fails to recognize that such an action would violate a posted Radiation Boundary and lead to potential uncontrolled dose.

'B' is wrong: Direct the EO to enter the area and evacuate workers. No such immediate requirement exists. Plausible to the examinee who fails to recognize that such an action would violate a posted Radiation Boundary and lead to potential uncontrolled dose.

'C' is wrong: Withdraw the 2E TIP into its shield. No such immediate requirement exists. Plausible to the examinee who fails to recognize that such an action would move the irradiated TIP detector through a normally accessible area.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 56 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	3.00	
System ID:	1104787	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LLOT1760.2	
Topic:	Determine required action to minimize dose when TIP is in operation	
Num Field 1:	3.4	
Num Field 2:	3.8	
Text Field:	2.3.14	
Comments:	Level	RO
	Tier	3*
	Group	N/A
	KA # and Rating	2.3.14 (3.4/3.8)
	KA Statement	2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.
	References	S74.0.A, Rev.56 RP-AA-460, Rev.26
	Examinee References	None
	Learning Objective	LLOT1760.2
	Question source	LGS 2012 ILT NRC Exam Question #72
	Question history	LGS 2012 ILT NRC Exam Question #72
	Cognitive level	Higher**
	10 CFR 55	41.12

	Comments	*Justification for Tier 3 item: Although the information regarding this question is found only in the TIP system operating procedure (S74.0.A), this item is nonetheless a legitimate Tier 3 question (i.e., not an "extension of Tier 2, per the requirements of NUREG-1021, Rev.9, Supp 1, section ES-401, page 6 of 33, section 2.a.) **Justification for Higher Cognitive categorization: Because the correct answer to this question is not explicitly stated in the S74.0.A procedure, but instead must be deduced from comprehending the intent of the Step 4.2 WARNING, this item is classified as HIGHER COGNITIVE. --- Editorial changes (to answer choices) to address NRC Rev.00 comments.

57 **ID: 1103428** **Points: 1.00**

Unit 1 is in a Station Blackout, with the following:

- HPCI and RCIC initiated on RPV low level
- RPV level is recovering
- E-1 (Loss of All AC Power (Station Blackout)) and T-101 are being executed concurrently

WHICH ONE of the following describes the required operator action per E-1?

A. Shutdown HPCI within 10 minutes of Station Blackout;
Transfer and maintain RCIC Pump suction to the Suppression Pool

B. Shutdown RCIC within 10 minutes of Station Blackout;
Transfer and maintain HPCI Pump suction to the Suppression Pool

C. Shutdown HPCI within 10 minutes of Station Blackout;
Maintain RCIC Pump suction aligned to the CST

D. Shutdown RCIC within 10 minutes of Station Blackout;
Maintain HPCI Pump suction aligned to the CST

Answer: A

Answer Explanation

E-1, step 2.1 directs operators to enter T-100/T-101 (RPV Control), as applicable, and execute it concurrently. Per step 3.1, if HPCI is automatically initiated, then HPCI shutdown per S55.2.A is to be completed within 10 minutes of the Station Blackout. The Limerick design basis for RPV water level control following a Station Blackout credits only the RCIC system for RPV level control since RCIC has sufficient capacity to maintain RPV inventory and HPCI capacity would result in exceeding the High RPV water level trip of +54 inches. Performance of S55.2.A returns the HPCI system to the auto/standby condition if the system has automatically initiated. Step 3.2 of E-1 provides direction to transfer and maintain RCIC suction to the Suppression Pool. The Limerick design basis for RPV level control for the four hour coping period following a Station Blackout credits the RCIC system in operation with suction from the Suppression Pool only. No credit is taken for the CST as a suction source for RCIC.

'A' is correct: Shutdown HPCI within 10 minutes of Station Blackout; Transfer and maintain RCIC Pump suction to the Suppression Pool. Correct for the reasons described above.

'B' is wrong: Shutdown RCIC within 10 minutes of Station Blackout; Transfer and maintain HPCI Pump suction to the Suppression Pool. Plausible to the examinee who cannot recall or is unfamiliar with (1) E-1 requirements for RPV water level control following a Station Blackout when E-1 and T-100/T-101 are being executed concurrently, and/or (2) the Limerick design basis requirements for RPV water level control following a Station Blackout. Also plausible in that HPCI and RCIC share the same suction sources.

'C' is wrong: Shutdown HPCI within 10 minutes of Station Blackout; Maintain RCIC Pump suction aligned to the CST. Plausible to the examinee who is unable to recall or is unfamiliar with (1) E-1 requirements for RPV water level control following a Station Blackout when E-1 and T-100/T-101 are being executed concurrently, and/or (2) the Limerick design basis requirements for RPV water level control following a Station Blackout. Also plausible in that HPCI and RCIC share the same suction sources.

'D' is wrong: Shutdown RCIC within 10 minutes of Station Blackout; Maintain HPCI Pump suction aligned to the CST. Plausible to the examinee who is unable to recall or is unfamiliar with (1) E-1 requirements for RPV water level control following a Station Blackout when E-1 and T-100/T-101 are being executed concurrently, and/or (2) the Limerick design basis requirements for RPV water level control following a Station Blackout. Also plausible in that HPCI and RCIC share the same suction sources.

Question 57 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.50	
System ID:	1103428	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	LGSOPS2000	
Topic:	E-1 Station Blackout - Recall operator actions	
Num Field 1:	3.8	
Num Field 2:	4.5	
Text Field:	2.4.8	
Comments:	Level	RO
	Tier	3
	Group	N/A
	KA # and Rating	2.4.8 (3.8/4.5)
	KA Statement	2.4.8 Knowledge of how abnormal operating procedures are used in conjunction with EOPs.
	References	E-1, Rev.45 E-1 Bases, Rev.9
	Examinee References	None
	Learning Objective	LGSOPS2000, no specified objective
	Question source	Bank 1101097
	Question history	LGS 2012 NRC Exam Question #73
	Cognitive level	Lower
	10 CFR 55	41.10
	Comments	None

58 **ID: 1118351** **Points: 1.00**

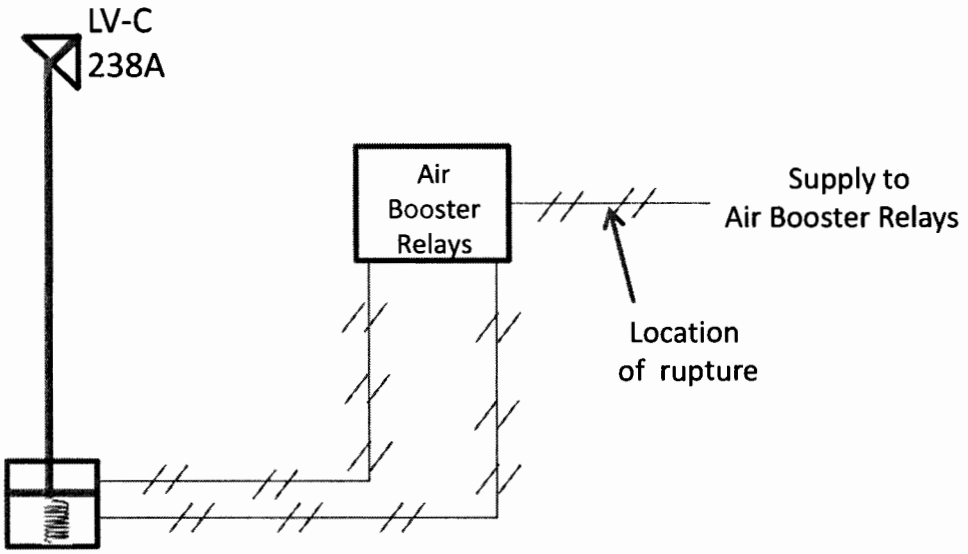
Unit 2 reactor startup is in progress, with the following:

- RPV level is steady
- Reactor pressure is 500 psig
- Feedwater to the RPV is through Startup Bypass Valve LV-C-06-238A
- LIC-006-238A is in AUTO; its "Output Demand" is 15%
- RFP Bypass Valve controller LIC-006-220 is in AUTO

The Instrument Air line supplying LV-C-06-238A ruptures (as shown on the figure below).

- Only the 238A valve is impacted

WHICH ONE of the following describes the response of RPV level **without** operator action?



A. Remains steady for a limited time then steadily rises

B. Begins to lower but then rises

C. Remains steady for a limited time then steadily lowers

D. Begins to rise but then lowers

Answer: C

Answer Explanation

While LV-C-06-238A is a fail-CLOSED valve (see P&ID M-0006, Sht.6, coordinate F-8, and NOTE 19 on that same drawing), its actuator is the type that "traps" operating air for a limited time before the air eventually bleeds off (leakage) and the valve fails CLOSED***. Unless another pathway for feeding the RPV becomes available, RPV level will respond as follows: remains steady for a limited time (i.e., the amount of time that the "238A" remains as-is due to the trapped air), then begins to steadily lower (i.e., as a result of 238A failing CLOSED).

[***NOTE - The embedded stem figure includes "Air Booster Relays"; these relays (one in each air output line going to the piston-type valve actuator. They are normally-open and air-operated. On a loss of air to them (such as a rupture at the location shown in the figure), the relays re-position, trapping the air supply in each of the two lines between them and the valve actuator.]

Stem conditions indicate the following key details:

- RPV pressure is only 500 psig (well below the Condensate Pumps' discharge pressure of approximately 600 psig), meaning that, given a pathway, the Condensate Pumps are able to feed the RPV and recover/maintain RPV level. [Refer to Condensate lesson plan, pages 2-3.]

- LIC-006-238A is in AUTO with an "Output Demand" signal of 15%

- RFP Bypass Valve (HV-006-220) controller LIC-006-220 is in AUTO

With LIC-006-220 in AUTO (as it is here), the HV-006-220 valve will automatically go into service (i.e., opening a pathway for feeding the RPV directly from the discharge of the Condensate Pump(s), bypassing the in-service RFP), so long as: the 238A controller "Output Demand" signal is <17% AND RPV pressure is <390 psig. With RPV pressure at 500 psig, the interlock is not satisfied; therefore, the 220 valve will not automatically open (although the RO could in fact force it open at MCR panel 20C603). [Refer to FWLCS lesson plan, pages 23-24.]

'C' is correct: Remains steady for a limited time then steadily lowers. Correct for the reasons described above.

'A' is wrong: Remains steady for a limited time then steadily rises. Plausible to the examinee who believes that LV-C-06-238A fails OPEN.

'B' is wrong: Begins to lower but then rises. Plausible to examinee who, seeing the 238A controller out at <17% and the 220 controller in AUTO, fails to consider that RPV pressure is too high (i.e., >390 psig) to satisfy the interlock that would otherwise automatically place the 220 valve in service. That examinee is especially distracted by the fact that RPV pressure (at 500 psig) is well within the discharge pressure of the Condensate Pumps; so, that examinee's mind-set is to quickly conclude that RPV level may well lower for a bit (only for as long as it takes for the transition from a failing-closed 238A valve to an automatically opening 220 valve), but then recover as the 220 valve begins to effectively feed the RPV.

'D' is wrong: Begins to rise but then lowers. Plausible to the examinee who believes that the 238A valve fails OPEN and also concludes (incorrectly) that the 220 valve will automatically go into service. Because the 220 valve has a smaller capacity (about 400-600 gpm) than does the 238A valve, it makes sense to that examinee that RPV level will peak but then begin to lower due to the reduced feed rate.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 58 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	3.00	
System ID:	1118351	
User-Defined ID:	REV 02, 01/06/15	
Cross Reference Number:	LLOT0540.14A	
Topic:	Predict impact of loss of Instrument Air Loss to LV-C-06-238A	
Num Field 1:	3.2	
Num Field 2:	3.3	
Text Field:	295019 AK2.03	
Comments:	Level	RO
	Tier	1
	Group	1
	KA # and Rating	295019 AK2.03 (3.2/3.3)
	KA Statement	295019 Partial or Total Loss of Instrument Air Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: AK2.03 Reactor feedwater
	References	P&ID M-0006, Sheet 6, Rev.25 LLOT0540, Feedwater System lesson plan LLOT0550, FWLCS lesson plan LGSOPS0005, Condensate System lesson plan
	Examinee References	None
	Learning Objective	LLOT0540.14A
	Question source	New*
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.7
	Comments	*Replacement "new" question (old ID: 1103429) to address NRC Rev.00 comments. To support this new item, added three lesson plans (in the "Added Exam References" folder that is part of this Exam Submittal). Editorial (added embedded figure to stem) to address NRC Rev.01 comments.

59 **ID: 1103430** **Points: 1.00**

Unit 1 plant conditions:

- Reactor is SHUTDOWN
- Suppression Pool level is 16.9'
- RCIC is injecting to the RPV with a suction temperature of 106°F
- 1A RHR is in Suppression Pool Cooling with a suction temperature of 115°F
- 1B RHR is secured; its suction temperature is 97°F
- SPOTMOS indicated temperature is 128°F
- CST level is 33.5'

WHICH ONE of the following identifies the **valid** Suppression Pool temperature, per T-102 (Primary Containment Control)?

A. 97°F

B. 106°F

C. 115°F

D. 128°F

Answer: C

Answer Explanation

Refer to T-102, NOTE #2, which reminds operators that the SPOTMOS probes are located in the suppression pool at an elevation which corresponds to an indicated suppression pool level of 17.8 ft. If indicated suppression pool level drops below 17.8 ft., Residual Heat Removal (RHR) pump suction temperature can be used as a valid alternate method for determining suppression pool temperature provided an RHR pump is running.

'C' is correct: 115°F. Correct for the reasons described above.

'A' is wrong: 97°F. Plausible to the examinee who recognizes that the supp pool level is too low to use SPOTMOS, but who mistakenly believes that the RHR suction temperature to be used is the one for an idle pump, rather than a running one.

'B' is wrong: 106°F. Plausible to the examinee who recognizes that the supp pool level is too low to use SPOTMOS, but who mistakenly believes that the RCIC pump suction temperature is to be used, rather than the running RHR pump suction temperature.

'D' is wrong: 128°F. Plausible to the examinee who either doesn't recognize the low supp pool water level or who simply fails to recall the T-102 Note #2.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 59 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	2.50	
System ID:	1103430	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	LLOT1560.5	
Topic:	T-102 - Identify the Supp Pool temperature with Supp Pool level is 16.9'	
Num Field 1:	3.9	
Num Field 2:	3.9	
Text Field:	295026 EA1.03	
Comments:	Level	RO
	Tier	1
	Group	1
	KA # and Rating	295026 EA1.03 (3.9/3.9)
	KA Statement	295026 Suppression Pool High Water Temp Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: EA1.03 Temperature monitoring
	References	T-102, Rev.24
	Examinee References	None
	Learning Objective	LLOT1560.5
	Question source	Bank 560663
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.7, 41.10
	Comments	None

60 **ID: 1118410** **Points: 1.00**

Consider the following types of devices installed in the Unit 1 RWCU Pump Room:

- Steam Flooding Damper
- Blowout Panel

WHICH ONE of the following identifies:

(1) the device that is designed to actuate at 0.5 psid, and
(2) the specific purpose for which <u>that</u> device is designed?

A. (1) Blowout Panel
(2) Prevent ventilation systems from rendering equipment in other rooms inoperable on high energy line break in RWCU Pump room

B. (1) Steam Flooding Dampers
(2) Prevent ventilation systems from rendering equipment in other rooms inoperable on high energy line break in RWCU Pump room

C. (1) Blowout Panel
(2) Relieve over-pressure from within RWCU Pump room in the event a high-energy pipe break in that room

D. (1) Steam Flooding Dampers
(2) Relieve over-pressure from within RWCU Pump room in the event a high-energy pipe break in that room

Answer: C

Answer Explanation

Generally, there are two types of components that are designed to mitigate an overpressure condition within a room/area: Blowout panels, and Steam Flooding Dampers (SFDs). Blowout panels are mounted only in the following rooms: Outboard MSIV Room, RWCU Pump Room, and HPCI and RCIC Pump Rooms, all of which are in Secondary Containment. Steam Flooding Dampers are located in many areas, all of which are also within Secondary Containment (refer to P&ID M-0076, Sheet 5, Table J, for a complete listing of the SFDs within the Unit 1 Reactor Enclosure).

Although both types of components do in fact "mitigate" an overpressure condition, only the blowout panels actually "relieve" pressure from within the room/area; i.e., the panels are installed as part of the room/area walls using differential-pressure sensitive explosive washers that act to disengage the panel from the wall upon sensing a fairly small d/p within that room/area. For example: the Unit 1 RWCU Pump Room has one blowout, mounted on the West wall of the Unit 1 Reactor Enclosure, which actuates at 0.5 psid to rapidly relieve pressure from within the RWCU Pump Room (due to a high-energy pipe break in the room) directly to the outside atmosphere. [Refer to Secondary Containment lesson plan, page 5-6 for a description of blowout panel operation, including actuation setpoints.]

Although Steam Flooding Dampers (SFDs) act to mitigate a room's high pressure condition, they do not "relieve" pressure from within that room. Rather, SFDs are installed in the HVAC supply and exhaust ducts for that room. They are normally-open and will automatically close when pressure inside the duct is 5" W.G. higher than pressure outside the duct (i.e., pressure within that room). By closing, the SFD simply prevents the high-energy pipe break problem in that room from propogating to adjacent Secondary Containment rooms/areas. [Refer to R.E. HVAC lesson plan, page 18 for a description of SFD operation and actuation setpoint.]

'C' is correct: (1) Blowout Panel; (2) Relieve over-pressure from within RWCU Pump room in the event a high-energy pipe break in that room. Correct for the reasons described above.

'A' is wrong: (1) Blowout Panel; (2) Prevent ventilation systems from rendering equipment in other rooms inoperable on high energy line break in RWCU Pump room. Part (2) is plausible to the examinee who fails to recognize that this is just another way to describe the function of a steam flooding damper; a blowout panel does not provide such a function.

'B' is wrong: (1) Steam Flooding Dampers; (2) Prevent ventilation systems from rendering equipment in other rooms inoperable on high energy line break in RWCU Pump room. This choice would be correct if the stem were asking about a device that actuates at 5" W.G., rather than at 0.5 psid. Plausible to the examinee who confuses the two actuation setpoints.

'D' is wrong: (1) Steam Flooding Dampers; (2) Relieve over-pressure from within RWCU Pump room in the event a high-energy. Plausible to the examinee who fails to recall the relative actuation setpoints for the two devices and also neglects to recognize that the SFD would infact isolate the room rather than allow it to propogate.

Question 60 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.50
System ID:	1118410
User-Defined ID:	REV 02, 01/06/15
Cross Reference Number:	LLOT0076B.6F
Topic:	Recall operation of Secondary Containment blowout panels and their purpose
Num Field 1:	2.8
Num Field 2:	3.1
Text Field:	295035 EK3.01
Comments:	Level RO Tier 1 Group 2 KA # and Rating 295035 EK3.01 (2.8/3.1)

	KA Statement	295035 Secondary Containment High Differential Pressure Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: EK3.01 Blow-out panel operation: Plant-Specific
	References	P&ID M-0076, Sheet 5, Rev.41 LLOT0076A, R.E. HVAC lesson plan LLOT0076B, Secondary Containment lesson plan
	Examinee References	
	Learning Objective	LLOT0076B.6F
	Question source	New*
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.5
	Comments	*Replacement "new" question (old ID: 1103431) to address NRC Rev.00 comments. Minor change (Part (2) of 'A' and 'B') to improve plausibility, to address NRC Rev.01 comments.

61 **ID: 1103474** **Points: 1.00**

Unit 1 LOCA is in progress.

Operators are entering T-112 (Emergency Blowdown) from Step PC/P-11 of T-102.

WHICH ONE of the following describes the specific reason for performing this Emergency Blowdown?

A. Ensure that primary containment vent valve operability is maintained

B. Ensure that the primary containment negative design pressure is not exceeded

C. Ensure that the SRV discharge lines are not damaged

D. Ensure that the pressure suppression function of the primary containment is maintained

Answer: D

Answer Explanation

Refer to T-102, specifically, the PC/P (Primary Containment Pressure) leg. Only when operators determine that the SAFE side of the Pressure Suppression Pressure (PSP) Curve, PC/P-3, cannot be maintained does Step PC/P-11 direct operators to perform a T-112 Emergency Blowdown. The T-102 Bases for Step PC/P-11 describes the PSP purpose as "to assure the pressure suppression function of primary containment is maintained...".

'D' is correct: Ensure that the pressure suppression function of the primary containment is maintained. Correct for the reasons described above.

'A' is wrong: Ensure that primary containment vent valve operability is maintained. This choice suggests the basis for Step PC/P-13...i.e., to vent the PC when it's determiend that pressure cannot be maintained below the Primary Containment Pressure Limit (PCPL) of Curve PC/P-1 (see the T-102 Bases for this step). Plausible to the examinee who mistakenly believes that this is the point in the PC/P leg where an Emergency Blowdown is preformed.

'B' is wrong: Ensure that the primary containment negative design pressure is not exceeded. This choice suggests the basis for Step PC/P-8...i.e., to terminate drywell sprays before drywell pressure drops to 0 psig (see the T-102 Bases for this step). Plausible to the examinee who confuses the two steps.

'C' is wrong: Ensure that the SRV discharge lines are not damaged. This choice suggests the basis for Step SP/l-19...i.e., to perform a T-112 Emergency Blowdown when it's determined that the SAFE side of the SRV Tailpipe Level Limit, Curve SP/L-1, cannot be restored and maintained. Plausible to the examinee who forgets that a concern for SRV tailpipe damage is found only in the SP/L leg strategy (see the T-102 basis for Step SP/L-19).

NOTE - At LGS, recall of EOP bases is RO-level knowledge supported by learning objectives.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 61 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	2.50	
System ID:	1103474	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LLOT1560.1	
Topic:	Recall reason for Emergency Blowdown based on PSP	
Num Field 1:	3.7	
Num Field 2:	4.1	
Text Field:	295024 EK3.04	
Comments:	Level	RO
	Tier	1
	Group	1
	KA # and Rating	295024 EK3.04 (3.7/4.1)
	KA Statement	295024 High Drywell Presure Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: EK3.04 †Emergency depressurization
	References	T-102, Rev.24 T-102 Bases, Rev.24
	Examinee References	None
	Learning Objective	LLOT1560.1
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.5, 41.10
	Comments	Editorial change to "Explanation" to address NRC Rev.00 comments.

62 **ID: 1103788** **Points: 1.00**

Plant conditions:

- Reactor scrammed (all rods in)
- Reactor pressure is 600 psig
- Reactor level is -145", down slow
- Suppression Pool level is 17'10"
- HPCI is running and injecting at 5600 gpm
- HPCI suction is aligned to the Suppression Pool
- No other RPV injection sources are available

WHICH ONE of the following describes the further operation of HPCI?

A. Can continue to inject at rated flow

B. Can continue to inject but only at reduced flow

C. HPCI suction must be transferred to the CST

D. HPCI operation must be terminated

Answer: D

Answer Explanation

Per T-102, Step SP/L-4, HPCI must be secured, regardless of adequate core cooling, if Suppression Pool level drops below 18'. Per the T-102 Bases for Step SP/L-4..."Operation of HPCI with its exhaust discharge not submerged will directly pressurize the Suppression Pool. The consequence of not securing HPCI may cause failure of the primary containment from overpressurization. Therefore, HPCI is secured regardless of adequate core cooling if Suppression Pool level drops to below 18'."

'D' is correct: HPCI operation must be terminated. Correct for the reasons described above.

'A' is wrong: Can continue to inject at rated flow. Plausible to the examinee who forgets the T-102 Step and its bases, or who does recall it ,except for the fact that the Step disregards the concern for adequate core cooling. This examinee is lulled into thinking this way by the fact that the stem conditions state that "no other RPV injection sources are available."

'B' is wrong: Can continue to inject but only at reduced flow. Plausible for reasons similar to that for choice 'A'. In this case, the examinee attempts to "rationalize" that it would be acceptable to continue operating HPCI at a reduced injection rate because that reduced rate would slow the rate at which the HPCI exhaust would pressurize the Suppression Pool. Again, the fact that "no other RPV injection sources are available" is highly distracting.

'C' is wrong: HPCI suction must be transferred to the CST. Plausible to the weaker examinee who completely forgets the concern for over-pressurizing the primary containment with the HPCI exhaust; this examinee instead focuses on whether or not the low pool level can provide adequate NPSH for the HPCI pump and concludes that it cannot, thereby warranting tranferring its suction to the CST.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 62 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	2.50
System ID:	1103788
User-Defined ID:	REV 01, 12/15/14
Cross Reference Number:	LLOT1560.5
Topic:	T-102 - Determine actions required for HPCI with low Supp Pool level
Num Field 1:	4.1
Num Field 2:	4.2
Text Field:	295030 EA2.01
Comments:	Level: RO Tier: 1 Group: 1 KA # and Rating: 295030 EA2.01 (4.1/4.2) KA Statement: 295030 Low Suppression Pool Water Level Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: EA2.01 Suppression pool level References: T-102, Rev.24 Examinee References: None Learning Objective: LLOT1560.5 Question source: Bank 560396 Question history: None Cognitive level: Higher 10 CFR 55: 41.10 Comments: Editorial change (answer choices) to address NRC Rev.00 comments.

63 **ID: 1103809** **Points: 1.00**

Unit 1 is operating at 100% power when the following occurs:

- RWCU demin resin spills during a resin transfer on R.E. elevation 283'
- All Reactor Enclosure HVAC Exhaust Rad Monitors indicate 1.2 mR/hr, steady

WHICH ONE of the following identifies the RMDS radiation monitor display that shoud be used to determine the offsite release rate?

A. WRAM Total Effluent

B. North Stack Normal Range

C. WRAM Low Range

D. South Stack Normal Range

Answer: D

Answer Explanation

Refer to P&ID M-0026, Sheet 1, to validate the following...

The NORTH STACK receives exhausts from the following:

- Unit 1 (2) Turbine Enclosures
- Radwaste Enclosure
- Unit 1 (2) Battery Rooms
- Unit 1 (2) Steam Packing Condensers
- Standby Gas Treatment System (SGTS)
- Unit 1 (2) Offgas

The SOUTH STACK receives exhausts from the following:

- Unit 1 (2) Reactor Enclosures
- Unit 1 (2) Refueling Floor

Normally, Reactor Enclosure (RE) HVAC is in operation, exhausting through the South Stack. If RE Ventilation Exhaust radiation rises to 1.35 mR/h4, RE HVAC isolates and SGTS initiates, then exhausting the Reactor Enclosure atmosphere through SGTS and ultimately through the North Stack. [Refer to alarm response card ARC-MCR-109, E1, to validate the isolation and setpoint.]

Stem condition indicate that the resin spill has caused RE Ventilation Exhaust to reach 1.2 mR/hr, which is below the isolation setpoint. Therefore, only the Radiation Monitor Display System (RMDS) displays associated with the South Stack will show any evidence of an offsite release resulting from the resin spill.

The Wide Range Accident Monitor (WRAM) channels (WRAM Total Effluent and WRAM Low Range) are associated only with the North Stack; the South Stack has no such instruments. Only the South Stack Normal Range channel will provide personnel with information related to the offsite release resulting from the resin spill. [Refer to P&ID M-0026, Sheet 6, to validate these instruments.]

'D' is correct: South Stack Normal Range. Correct for the reasons described above.

'A' is wrong: WRAM Total Effluent. Plausible to the examinee who does not recognize that the RE HVAC rad levels of 1.2 mR/hr were NOT high enough to cause the isolation. Thus, that examinee believes the SGTS went into service and is exhausting through the North Stack. Also plausible to an examinee does recognize that ventilation is still exhausting through the South Stack, but incorrectly believe that the South Stack has a WRAM monitor; it does not.

'B' is wrong: North Stack Normal Range. Plausible for reasons similar to that for choice 'A'.

'C' is wrong: WRAM Low Range. Plausible for reasons similar to that for choices 'A' and 'B'.

Question 63 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	2.50
System ID:	1103809
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LGSOPS0026B.4
Topic:	Determine RMDS display to use for offsite release
Num Field 1:	3.9
Num Field 2:	4.2
Text Field:	295038 EA1.01
Comments:	Level: RO Tier: 1 Group: 1 KA # and Rating: 295038 EA1.01 (3.9/4.2) KA Statement: 295038 High Offsite Release Rate Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: EA1.01 Stack-gas monitoring system References: P&ID M-0026, Sheet 1, Rev.37 P&ID M-0026, Sheet 6, Rev.29 ARC-MCR-109, E1, Rev.2 Examinee References: None Learning Objective: LGSOPS0026B.4 Question source: New Question history: None Cognitive level: Higher 10 CFR 55: 41.7 Comments: None

64 **ID: 1103884** **Points: 1.00**

Unit 1 operators are executing T-103 (Secondary Containment Control) for a primary system discharge into Secondary Containment.

Per T-103, the operators decide to re-align the discharge of the Reactor Enclosure Floor Drain Sump, per T-236, in order to reduce the offsite release rate.

WHICH ONE of the following describes the flowpath for the re-aligned floor drain sump discharge?

To the Suppression Pool via...

A. Core Spray suction piping

B. RHR Loop 'A' suction piping

C. RHR Loop 'B' suction piping

D. RCIC suction piping

Answer: A

Answer Explanation

T-103, Step SCC/L-5 permits the re-alignment of the R.E. floor drain sump per T-236. T-236 shuts off the sump discharge to DRW and opens a discharge flowpath to the suppression pool via the Core Spray suction piping.

'A' is correct: To the Suppression Pool via Core Spray suction piping. Correct for the reasons described above.

'B' is wrong: To the Suppression Pool via RHR Loop 'A' suction piping. Plausible for two reasons: (1) T-236 is a seldom, if ever, part of the ILT simulator phase of training; it is therefore likely to be unfamiliar to most ILT Candidates; (2) as such, the suggestion that the flowpath would be via an ECCS system other than Core Spray is quite believable.

'C' is wrong: To the Suppression Pool via RHR Loop 'B' suction piping. Plausible for two reasons: (1) T-236 is a seldom, if ever, part of the ILT simulator phase of training; it is therefore likely to be unfamiliar to most ILT Candidates; (2) as such, the suggestion that the flowpath would be via an ECCS system other than Core Spray is quite believable.

'D' is wrong: To the Suppression Pool via RCIC suction piping. Plausible for two reasons: (1) T-236 is a seldom, if ever, part of the ILT simulator phase of training; it is therefore likely to be unfamiliar to most ILT Candidates; (2) as such, the suggestion that the flowpath would be via an ECCS system other than Core Spray is quite believable.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 64 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.00	
System ID:	1103884	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LGSOPS2003.IL4	
Topic:	Recall T-236 flowpath for realigning RE Floor Drain Sump discharge	
Num Field 1:	3.1	
Num Field 2:	3.2	
Text Field:	295036 EK2.01	
Comments:	Level	RO
	Tier	1
	Group	2
	KA # and Rating	295036 EK2.01 (3.1/3.2)*
	KA Statement	295036 Secondary Containment High Sump/Area Water Level Knowledge of the interrelations between SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL and the following: EK2.01 Secondary containment equipment and floor drain system
	References	T-103, Rev.20 T-236 U/1, Rev.14
	Examinee References	None
	Learning Objective	LGSOPS2003.IL4
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.10

	Comments	*Justification for KA match: Exam Author, Facility Rep, and Exelon OTPS concensus that this KA (295036 EK2.01) leaves a wide-open door to most any question that ties Secondary Containment High Water Level to the Secondary Containment equipment/floor drain system. LGS has always perceived the 295xxx K2's as being treated much the same as SYSTEM K1's. We believe this question represents a solid KA match. Nonetheless, made an editorial change to the stem and answer choices to be consistent with other items on the RO exam that were similarly changed to address NRC Rev.00 comments.

65 **ID: 1103891** **Points: 1.00**

Operators are executing T-117, Level/Power Control.

WHICH ONE of the following describes the **EARLIEST** point in T-117 execution when it is permissible to inject to the RPV with HPCI via Core Spray?

A. First Terminate/Prevent has been performed; the target RPV level band is -100" to -60"

B. Second Terminate/Prevent has been performed; the target RPV level band is now -150" to -110"

C. Emergency Blowdown has not been performed; RPV level is -190" and lowering with all other available systems injecting

D. Emergency Blowdown has been performed; RPV level is -190" and lowering with all other available systems injecting

Answer: D

Answer Explanation

Refer to T-117, Step LQ-25...this is the EARLIEST point at which injection inside the core shroud is permitted. This step comes only after T-112 (Emergency Blowdown) has been performed AND it's determined that level cannot be restored and maintained above -186".

'D' is correct: Emergency Blowdown has been performed; RPV level is -190" and lowering with all other available systems injecting. Correct for the reason described above.

'A' is wrong: First Terminate/Prevent has been performed; the target RPV level band is -100" to -60".

'B' is wrong: Second Terminate/Prevent has been performed; the target RPV level band is now -150" to -110".

'C' is wrong: Emergency Blowdown has not been performed; RPV level is -190" and lowering with all other available systems injecting.

Distracters are all plausible because they cite the several other times during the execution of T-117, specifically Steps LQ-16, 17, and 22, when RPV re-injection is directed.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 65 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	1103891
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LLOT1560.5
Topic:	T-117 - Recall EARLIEST point at which injection inside the core shroud is permitted
Num Field 1:	3.7
Num Field 2:	4.7
Text Field:	206000 2.4.6
Comments:	Level RO Tier 2 Group 1 KA # and Rating 206000 2.4.6 (3.7/4.7) KA Statement 206000 HPCI 2.4.6 Knowledge of EOP mitigation strategies. References T-117, Rev.17 Examinee References None Learning Objective LLOT1560.5 Question source Bank 989228 Question history None Cognitive level Lower 10 CFR 55 41.10 Comments None

66 **ID: 1103894** **Points: 1.00**

Unit 1 plant startup is in progress after a forced outage, with the following:

- The date is August 15

The PRO has been directed to perform the following at 10C655:

- Place the "preferred" two Service Water (SW) Pumps in service per S10.1.A (SW System Startup and Normal Operation)

1B SW Pump is now operating.

WHICH ONE of the following identifies:

(1) the next SW Pump to be started, and
(2) the parameter available to the PRO (at 10C655) to confirm that pump flows are in the acceptable range?

A. (1) 1C or 1A
(2) motor current

B. (1) 1C only
(2) motor current

C. (1) 1C only
(2) pump d/p

D. (1) 1C or 1A
(2) pump d/p

Answer: B

Answer Explanation

Refer to S10.6.A (Swapping SW Pumps), Section 3.5 and its NOTE for information concerning the preferred (most desired) SW Pump configuration for summer (3.5.1) and winter (3.5.2). Stem indicates the date is August 15; therefore, the examinee is expected to recognize that, with the 1B pump operating, the "next" pump to be started can only be 1C. If this were a winter evolution, then either the 1A or the 1C pump could be started next.

Refer to S10.7.C (SW Flow Adjustments), Section 3.0 NOTE, which explains how either pump d/p (psid), actual flow (gpm), or motor current (amps) may be used to confirm that pump flows are within their acceptable range. However, at LGS, only pump amps (motor current) is available in the MCR (at *0C655); the S10.7.C (and similar S10 procedures) provide specific ranges of motor current (amps) that represent acceptable pump flows, for both single-pump and two-pump operation.

'B' is correct: (1) 1C only; (2) motor current. Correct for the reasons described above.

'A' is wrong: (1) 1C or 1A; (2) motor current. Part (1) is plausible to the examinee who doesn't recall the summer/winter configurations.

'C' is wrong: (1) 1C only; (2) pump d/p. Part (2) is plausible to the examinee who doesn't recall the SW indication available in the MCR.

'D' is wrong: (1) 1C or 1A; (2) pump d/p. Plausible for the same reasons as that for choices 'A' and 'C'.

Question 66 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	2.50	
System ID:	1103894	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	LGSOPS0010.6	
Topic:	Recognize Preferred Service Water Lineup for Summer and Predict Motor Currents	
Num Field 1:	2.8	
Num Field 2:	2.8	
Text Field:	400000 A1.01	
Comments:	Level	RO
	Tier	2
	Group	1
	KA # and Rating	400000 A1.01 (2.8/2.8)
	KA Statement	400000 Component Cooling Water Ability to predict and / or monitor changes in parameters associated with operating the CCWS controls including: A1.01 CCW flow rate
	References	S10.6.A, Rev.21 S10.7.C, Rev.27
	Examinee References	None
	Learning Objective	LGSOPS0010.6
	Question source	New
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.5, 41.10
	Comments	None

67 **ID: 1103944** **Points: 1.00**

Unit 1 is operating at 100% power when the following occurs:

- All power is lost from 1BY185

Operators stabilize the plant.

Then:

- Due to toxic gas, operators perform the Immediate Actions of SE-1 (Remote Shutdown) and evacuate the MCR

Consider the following indications for determining that all control rods have fully inserted:

1. "FULL-IN" lights are lit on the FCD
2. "ALL RODS IN" status is shown on PMS
3. "Rods not full-in" LED is extinguished at the RDCS cabinet in the AER

WHICH ONE of the following identifies (from the above list) the indications, if any, that operators can use to confirm that all rods have fully inserted?

(Assume all light bulbs / LEDs are good.)

A. 1, 2, and 3

B. 2, only

C. 3, only

D. None

Answer: D

Answer Explanation

Refer to 1S73.1.B (COL), Electrical Panel Alignment for Reactor Manual Control System, page 3, Step 6, which shows that the 1BY185 (static inverter) powers the "RPIS 10C615" panel. This panel powers the circuits for all of the RPIS Position-Indicating-Probe (PIP) reed switches. Loss of 1BY185 results in completely disabling the PIP for all 185 control rods; i.e., RPIS is incapable of providing any control rod position indication that is reliant on functioning PIPs.

All three of the indications suggested in the stem conditions (i.e., the "Full-In" lights on the MCR Full Core Display, the "All Rods In" status message on the Plant Process Computer screem (PMS), and the "Rods Not Full In" LED on the Rod Drive Control System (RDCS) cabinet in the aux Equipment Room (AER) rely on functioning PIPs. Therefore, none of the three suggested indications can be used by the operators.

'D' is correct: None. Correct for reasons described above.

'A' is wrong: 1, 2, and 3. Plausible to the examinee who does recall that one of the APRM static inverters (1AY185 or 1BY185) powers the RPIS panel, but who mistakenly chooses the wrong inverter (1AY185). Thus, the conclusion that 1AY185 is still powered leads the examine to conclude that RPIS in unaffected.

'B' is wrong: 2, only. Plausible to the examinee who recognizes that the 1BY185 loss has affected RPIS, but who mistakenly believes that the plant process computer (PMS) does not rely on RPIS inputs; therefore, the "All Rods In" status message is still available to the operators.

'C' is wrong: 3, only. Plausible to the examinee who recognizes that the 1BY185 loss has affected RPIS, and therefore, has disabled both the FCD and the PMS indications. However, that examinee mistakenly believes that the RDCS (being the "heart" of RMCS) is somehow not reliant on the PIPs.

Question 67 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	1103944
User-Defined ID:	REV 01, 12/15/14
Cross Reference Number:	LGSOPS0073A.4
Topic:	Predict impact of 1BY185 loss on RPIS
Num Field 1:	2.7
Num Field 2:	2.8
Text Field:	214000 K5.01
Comments:	Level: RO Tier: 2 Group: 2 KA # and Rating: 214000 K5.01 (2.7/2.8) KA Statement: 214000 RPIS Knowledge of the operational implications of the following concepts as they apply to ROD POSITION INFORMATION SYSTEM: K5.01 Reed switches References: 1S73.1.B (COL), Rev.6 Examinee References: None Learning Objective: LGSOPS0073A.4 Question source: New Question history: None Cognitive level: Higher 10 CFR 55: 41.7 Comments: Editorial to stem to address NRC Rev.00 comments.

68 **ID: 1103946** **Points: 1.00**

Operators have evacuated the MCR due to toxic gas.

Per SE-1, all Remote Shutdown Transfer Switches (at the RSP) have been placed in EMERGENCY.

WHICH ONE of the following describes a manipulation that Unit 1 operators will have to perform in order to place '1A' RHR in Suppression Pool Cooling?

A. Throttle open the '1A' RHR Pump Full Flow Test Return (HV-51-1F024A) locally at the valve

B. Close the '1A' RHR HX Bypass (HV-51-1F048A) locally at the valve

C. After starting the '1A' RHR Pump, open the '1A' RHR Pump Min Flow Valve (HV-51-1F007A) at the RSP

D. Open the '1A' RHR HX Inlet (HV-51-1F047A) at the RSP

Answer: C

Answer Explanation

Refer to SE-1 (Remote Shutdown), Section 4.7 and Attachment 1.

Although the RHR HX Inlet (F047) does have a control switch at the RSP (Remote Shutdown Panel); the valve is normally-OPEN, therefore, no manipulation is required.

Although the RHR HX Bypass (F048) does have to be closed, it has a control switch at the RSP; therefore, there is no need to manually stroke it closed locally at the valve.

The Full Flow Test Return (F024) has a control switch at the RSP; therefore, there is no need to manually throttle it for the desired flow locally at the valve.

With all RST Switches having been placed in EMERGENCY, the auto-cycle feature for the RHR Pump Min Flow Valve (F007) is disabled; therefore, the operator is directed to immediately open the valve (control switch at the RSP) after starting the pump.

'C' is correct: After starting the '1A' RHR Pump, open the '1A' RHR Pump Min Flow Valve (HV-51-1F007A) at the RSP. Correct for the reasons described above.

'A' is wrong: Throttle open the '1A' RHR Pump Full Flow Test Return (HV-51-1F024A) locally at the valve. Plausible to the examinee who fails to recall that the F024 valve can be controlled at the RSP.

'B' is wrong: Close the '1A' RHR HX Bypass (HV-51-1F048A) locally at the valve. Plausible to the examinee who fails to recall that the F048 valve can be controlled at the RSP.

'D' is wrong: Open the '1A' RHR HX Inlet (HV-51-1F047A) at the RSP. Very plausible to the examinee who recognizes that the RTS Switches in EMERGENCY have disabled the auto capability of various valves, but who fails to recall that the F047 valve is, first of all, already normally-open, and that it has no automatic functionality.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 68 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	2.50	
System ID:	1103946	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	LLOT0051.12	
Topic:	Recall how to operate RHR in SP Cooling when in SE-1	
Num Field 1:	3.0	
Num Field 2:	3.2	
Text Field:	219000 K4.05	
Comments:	Level	RO
	Tier	2
	Group	2
	KA # and Rating	219000 K4.05 (3.0/3.2)
	KA Statement	219000 RHR/LPCI: Torus/Pool Cooling Knowledge of RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE design feature(s) and/or interlocks which provide for the following: K4.05 Pump minimum flow protection
	References	SE-1, Rev.70
	Examinee References	None
	Learning Objective	LLOT0051.12
	Question source	Bank 989411
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.7, 41.10
	Comments	None

69 **ID: 1103947** **Points: 1.00**

Unit 1 is operating at 100% power when the following occurs:

- Simultaneous "hard" failure of Steam Flow transmitters FT-041-1N003B and FT-041-1N003D

Shortly thereafter, a lowering main condenser vacuum forces operators to successfully insert a manual scram using RPS.

WHICH ONE of the following identifies:

(1) the mode of level control (Single-element/Three-element) after the steam flow transmitters fail, and
(2) the response of the RFPTs after the RPS actuation?

A. (1) Single-element
(2) Speeds immediately begin to ramp down to minimum speed

B. (1) Three-element
(2) Speeds remain as is for 10 seconds then begin to ramp down to establish 10% Total FW flow

C. (1) Single-element
(2) Speeds remain as is for 10 seconds then begin to ramp down to establish 10% Total FW flow

D. (1) Three-element
(2) Speeds immediately begin to ramp down to minimum speed

Answer: C

Answer Explanation

Refer to S06.1.H U/1, Attachment 1 Alarm List, specifically: Signal Identity 1XX-FW301.ISFE, found at the top of Attachment 1, page 10. There, we find that the simultaneous failure of FT-041-1N003B and FT-041-1N003D identifies a "Steam Flow SMS Error." The 'Automatic Actions' column reveals that FWLCS will auto-swap to Single-Element control; however, the 'Operator Actions' column does not indicate that the forced single-element control will prevent the Scram Profile from being activated. In fact, FWLCS will swap to Three-Element control and activate the Scram Profile when an RPS scram is initiated (this has been validated to be correct on the LGS Simulator). This "successful" activation of the Scram Profile, with a pre-existing dual-steam flow transmitter failure would be very different were the pre-existing failure a dual-feed flow transmitter failure for a single RFPT. In that case, the Scram Profile will NOT activate in response to the RPS scram actuation; as such, the RFPT speeds would immediately begin to ramp down toward their minimum speed (~2300 rpm)...refer to Signal Identity 1XX-FW302.ITFFE, on page 15 of the Attachment 1 Alarm List, to validate this claim.

Refer to S06.1.D U/1 (Post Scram Level Control), Attachment 1, page 1 of 4, especially Figure 1, for a description iof how the RFPTs respond when the Scram Profile is activated...namely, that the RFPT speeds lock in place (remain as is) for 10 seconds, after which they begin to ramp down so as to produce a controlled lowering of the total feedwater flow (at a rate 6% flow per second) until 10% TFW is established.

'C' is correct: (1) Single-element; (2) Speeds remain as is for 10 seconds then begin to ramp down to establish 10% Total FW flow. Correct for the reasons described above.

'A' is wrong: (1) Single-element; (2) Speeds immediately begin to ramp down to minimum speed. Part (2) is plausible to the examinee who confuses the impact of a dual-steam flow transmitter with that of a dual-feed flow transmitter failure (as described above).

'B' is wrong: (1) Three-element; (2) Speeds remain as is for 10 seconds then begin to ramp down to establish 10% Total FW flow. Part (1) is plausible to the examinee who recogizes that the Scram Profile does activate, but also recalls that FWLC must be in Three-element control in order for that to occur. As such, that examinee fails to carefully read the Part (1) of the question statement (which asks for the FWLC status after the dual-trasnmitter failure); the examinee inappropriately applies Part (1) to what he/she knows to be the FWLC status (Three-element) after the RPS actuation, instead.

'D' is wrong: (1) Three-element; (2) Speeds immediately begin to ramp down to minimum speed. Plausible for reasons similar to that for choices 'A' and 'B'.

Question 69 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	2.50	
System ID:	1103947	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LLOT0550,11	
Topic:	FWLCS Steam Flow Transmitter Failure - Predict RFPT response to RPS scram	
Num Field 1:	3.8	
Num Field 2:	3.8	
Text Field:	259001 K6.07	
Comments:	Level	RO
	Tier	2
	Group	2
	KA # and Rating	259001 K6.07 (3.8/3.8)
	KA Statement	259001 Reactor Feedwater Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR FEEDWATER SYSTEM: K6.07 Reactor water level control system
	References	S06.1.H U/1, Rev.14 S06.1.D U/1, Rev.23
	Examinee References	None
	Learning Objective	LLOT0550,11

	Question source	Modified Bank 946742 - Revised Part (2) of question statement and of each answer choice. Was originally phased as "Scram Profile Activiated, or Not"; is now phrased as "RFPT response after the RPS actuation". Modified in this way in order to more closely match the selected KA statement.
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.7
	Comments	Editorial to stem to address NRC Rev.00 comments.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

70 **ID: 1103964** **Points: 1.00**

Unit 1 is operating at 9% power during a GP-2, PLANT STARTUP, with the following:

- The RWM is currently latched to Group 7
- Group 7 INSERT limit is 8, and WITHDRAW limit is 24

A CRD flow controller fault results in the following:

- Control rod 10-27 is inadvertently withdrawn to position 30
- Reactor power rises to 12%

WHICH ONE of the following identifies the required operator action, per ON-123 (Mispositioned Control Rods), before contacting the R.E.?

A. First, bypass the RWM, then insert control rod 10-27 to position 24

B. Leave the RWM as is and restore control rod 10-27 to position 24

C. Leave the RWM as is, immediately correct the error and demand a P-1 edit

D. First, bypass the RWM, then insert control rod 10-27 to position 00

Answer: A

Answer Explanation

Upon entry into ON-123, examinee will review Steps 2.1 thru 2.4, then proceed to Step 2.5, where he/she should recognize, from the stem conditions, that control rod 10-27 is withdrawn beyond the "one notch" provision of Step 2.5. Within Step 2.5, the examinee recognizes that the action of the 1st bullet applies because the resulting 12% reactor power is in fact still below the LPSP (Low Power Setpoint). At LGS, the LPSP is 15.9% power based on Total Steam Flow. Below the LPSP, the Rod Worth Minimizer (RWM) enforces its rod blocks. The examinee is expected to recognize that the control rod 10-27 being greater than two notches past its Withdraw Limit (two notches would be Position 28; the rod is actually at Position 30), causes the RWM to generate both an INSERT and WITHDRAW rod block (of all control rods). Therefore, operators must manually bypass the RWM (done at the RWM itself) before they will be able to move control rod 10-27. Note - Although Step 2.6 does not explicity say "Bypass the RWM", the examinee is expected to recognize that, with the rod mispositioned by >2 notches, the RWM will have to first be bypassed, before any movement of the rod is possible.

'A' is correct: First, bypass the RWM, then insert control rod 10-27 to position 24. Correct for the reasons described above. This is taken from Step 2.5, 1st bullet.

'B' is wrong: Leave the RWM as is and restore control rod 10-27 to position 24. Plausible to the examinee who fails to recognize that the INSERT and WITHDRAW rod blocks are present. In fact, if control rod 10-27 had landed at Position 28 rather than Position 30, the RWM would not have to be bypassed; this answer choice would be correct. This is also taken from Step 2.5, 1st bullet, but, as discussed earlier, ignores the need to manually bypass the RWM.

'C' is wrong: Leave the RWM as is, immediately correct the error and demand a P-1 edit. This is taken from Step 2.5, 3rd bullet, which would apply if power were above the LPSP; plausible for the reasons already described above.

'D' is wrong: First, bypass the RWM, then insert control rod 10-27 to position 00. This is taken from Step 2.4, 1st bullet, which would apply if this event had occurred during a plant shutdown (i.e., while rods are being inserted). Plausible to the examinee who does recognize that power is below the LPSP, but who fails to recognize that Step 2.4 doesn't apply for this situation.

Question 70 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	5	
Difficulty:	2.00	
System ID:	1103964	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LGSOPS1550.3	
Topic:	Predict RWM response to mispositioned rod and determine ON-123 action to mitigate	
Num Field 1:	3.1	
Num Field 2:	3.5	
Text Field:	201006 A2.05	
Comments:	Level	RO
	Tier	2
	Group	2
	KA # and Rating	201006 A2.05 (3.1/3.5)
	KA Statement	201006 RWM Ability to (a) predict the impacts of the following on the ROD WORTH MINIMIZER SYSTEM (RWH) (PLANT SPECIFIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.05 Out of sequence rod movement
	References	ON-123, Rev.22
	Examinee References	ON-123 (entire proecedure)
	Learning Objective	LGSOPS1550.3
	Question source	Bank 976942
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.5, 41.10
	Comments	Editorial to stem and answer choices to address NRC Rev.00 comments.

71 **ID: 1103966** **Points: 1.00**

Unit 2 is operating at 100% power when operators insert a manual scram due to rapidly rising Drywell pressure.

An ATWS occurs:

- RPV level reaches -130" before RCIC stabilizes it there
- RPV pressure is 700 psig, steady

CRS directs operators to Terminate and Prevent RPV Injection using T-270.

WHICH ONE of the following describes the **REQUIRED** controls manipulation(s) (at 20C601) necessary to terminate/prevent either Core Spray or LPCI?

A. Place all 4 Core Spray Pump handswitches to STOP.
Place the handswitches to CLOSE and release for HV-52-2F005 and HV-52-2F037.

B. Place the handswitches to CLOSE (and release) for HV-52-2F005 and HV-52-2F037.
No additional manipluations are required for Core Spray.

C. Place all 4 RHR Pump handswitches to STOP.
No additional manipulations are required for RHR.

D. Place all 4 RHR Pump handswitches to STOP.
Place all 4 LPCI Injection Valve handswitches to CLOSE and release.

Answer: C

Answer Explanation

Refer to T-270, Section 4.7, which provides the direction for preventing the LOCA automatic start of the Core Spray and RHR Pumps, by positioning to "TEST" the S11 and S44 "Pump 4KV Bus Power Monitor" Switches. Refer to Core Spray Elementary Drawing E21-1040-E-005, Sheet 001...the S11A switch (for example) is shown at coordinate F-7; when the switch is placed in TEST, it de-energizes relay K10A, disabling the control circuit for the 'A' Core Spray Pump...coordinate D-7 shows a now-open contact T1-M1 for K10A, which de-energizes relays K25A and K26A...ccordinates C/D-4/5 show now-open contacts for K25A and K26A that disable the control circuit for the Core Spray Loop A Shutoff Valve 2F005. all of this simply shows that performing T-270 Section 4.7 results in overriding the ECCS pumps "OFF" and their associated injection/shutoff valves "CLOSED". Refer to RHR Elementary Drawing E11-1040-E-005, Sheet 001 to validate the same design for the RHR Pumps and LPCI Injection Valves (2F017's).

Given the sequence of events described in the question stem conditions, the examinee is expected to recognize that the LOCA signal occurred before the CRS called for perfromance of T-270.

Refer to T-270, Section 4.4, Core Spray Terminate/Prevent:

- CAUTION 2 of Step 4.4.1 reminds operators that if Section 4.7 wasn't performed before the LOCA signal (it wasn't) AND the Core Spray valves (2F005 and 2F037) have OPENED (they have not), then the operators must CLOSE those valves using their handswitches at 20C601.
- However, the examinee is expected to recognize that although the Core Spray pumps are running (LOCA signal), the 2F005 and 2F037 valves ares still CLOSED; the reason is that the valve control logic "opening permissive" requires RPV pressure to be less than 455 psig (stem conditions indicate RPV pressure is steady at 700 psig).

Refer to T-270, Section 4.5, RHR Terminate/Prevent:

- CAUTION 2 of Step 4.51 is essentially the same as that for Core Spray Section 4.4.
- Similar to Core Spray, the RHR pumps are also running, but their associated LPCI injection valves (2F017A, B, C, D) are still CLOSED; again, their "opening permissive" (<74 psid between RHR discharge pressure and RPV pressure) is not satisfied with a 700 psig RPV pressure.

Therefore, the **REQUIRED** controls manipulation(s) to terminate/prevent either of these two ECCS systems is simply to...STOP the running pumps. T-270, Step 4.1.1 will, in every case, dispatch an operator to the AER to perform Section 4.7, Until RPV pressure drops enough to satisfy the opening permissive for the injection/shutoff valves, they'll remain CLOSED. and once Section 4.7 is completed, they will remain overridden CLOSED regardless of RPV pressure.

'C' is correct: Place all 4 RHR Pump handswitches to STOP; No additional manipulations are required for RHR. Correct for the reasons described above.

'A' is wrong: Place all 4 Core Spray Pump handswitches to STOP; Place the handswitches to CLOSE and release for HV-52-2F005 and HV-52-2F037. Plausible to the examinee who fails to recognize that the 2F005 and 2f037 are still CLOSED (with RPV pressure steady at 700 psig). Also plausible to the examinee who does recognize that the 2F005 and 2F037 are still CLOSED, but mistakenly believes that the impending performance of Section 4.7 will only "override OFF" the pumps and not also the injection/shutoff valves. This is a common mistake because inexperienced operators/Candidates only recall the names of the S11 and S44 TEST switches as being "Pump 4KV Bus Power Monitor" switches.

'B' is wrong: Place the handswitches to CLOSE (and release) for HV-52-2F005 and HV-52-2F037; No additional manipluations are required for Core Spray. Plausible for reasons similar to that for choice 'A'. This examinee also fails to recognize that the pumps have in fact auto-started and so must be STOPPED using the handswitches.

'D' is wrong: Place all 4 RHR Pump handswitches to STOP; Place all 4 LPCI Injection Valve handswitches to CLOSE and release. Plausible for reasons similar to that for choice 'A'; except, in this case the incorrect thinking applies to RHR.

Question 71 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	2.50
System ID:	1103966
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	NONE
Topic:	Given an ATWS/LOCA, recognize the T-270 manips required to terminate/prevent Core Spray and RHR
Num Field 1:	4.6
Num Field 2:	4.6
Text Field:	203000 2.1.20
Comments:	Level RO Tier 2

	Group	1
	KA # and Rating	203000 2.1.20 (4.6/4.6)
	KA Statement	203000 RHR/LPCI: Injection Mode 2.1.20 Ability to interpret and execute procedure steps.
	References	T-270 Unit 2, Rev.13 Drawing E21-1040-E-005, Sheet 001, Rev.18 Drawing E11-1040-E-005, Sheet 001, Rev.26
	Examinee References	None
	Learning Objective	No specified objective
	Question source	New
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.7, 41.8, 41.10
	Comments	None

72 **ID: 1104004** **Points: 1.00**

Unit 1 is operating at 100% power when the following occurs:

- The "preferred" input source to 1AD185 is lost

WHICH ONE of the following identifies how the loads on 1AY185 are remaining energized?

A. From 1AY160 through the 1AD185 Static Switch

B. From 1AY160 bypassing the 1AD185 Static Switch

C. Directly from 1AD160 through the 1AD185 Static Switch

D. Directly from 1AD160 bypassing the 1AD185 Static Switch

Answer: A

Answer Explanation

Refer to electrical drawing E-0032, Sheet 2. When the normal "Preferred" input source (125 VDC from 1AD108 DC Distribution Panel) to the APRM UPS, 1AD185, is lost, the 1AD185 Static Switch rapidly transfers over to the "Alternate" input source, which is 120 VAC from the 'A' RPS/UPS Distribution Panel 1AY160.

'A' is correct: From 1AY160 through the 1AD185 Static Switch. Correct for the reasons described above.

'B' is wrong: From 1AY160 bypassing the 1AD185 Static Switch. Plausible to the examinee who forgets that the Static Switch is the vital component necessary for the 1AY185 loads to remain "uninterruptibly" energized when the 1AD185 "inverter" section fails to produce its normal 120 VAC output (because of the loss of the required 125 VDC input to the inverter section).

'C' is wrong: Directly from 1AD160 through the 1AD185 Static Switch. Plausible to the examinee who mistakenly believes that 1AD185 is a load directly off of the 'A' RPS/UPS (1AD160), rather than being simply just another load on the 1AY160 Distribution Panel.

'D' is wrong: Directly from 1AD160 bypassing the 1AD185 Static Switch. Plausible for the same reasons as that for choices 'B', 'C'.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 72 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.00	
System ID:	1104004	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	LGSOPS0094.7	
Topic:	Recall how 1AY185 loads remain energzied on 1AD185 loss of DC input	
Num Field 1:	2.8	
Num Field 2:	3.1	
Text Field:	262002 K6.02	
Comments:	Level	RO
	Tier	2
	Group	1
	KA # and Rating	262002 K6.02 (2.8/3.1)
	KA Statement	262002 UPS (AC/DC) Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.): K6.02 D.C. electrical power
	References	Drawing E-0032, Sheet 2, Rev.39
	Examinee References	None
	Learning Objective	LGSOPS0094.7
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.7
	Comments	None

73 **ID: 1104005** **Points: 1.00**

Unit 2 power ascension is in progress after completing 2R13.

A Special Test for the newly installed DEHC system is in progress, with the following:

- Load Limit, Load Set, Pressure Set, and MCFL are all set at their normal full power values

Reactor power is 80% when the PRO is directed to perform the following at the Bypass Valve (BPV) Jack screen:

- Press the FULL OPEN button, then
- Press the HOLD button 10 seconds later

WHICH ONE of the following describes:

(1) how the BPVs respond, and
(2) the effect on the CV flow demand signal?

A. (1) one or more have opened, but not all of them
(2) the signal has increased

B. (1) one or more have opened, but not all of them
(2) the signal has lowered

C. (1) all have partially opened
(2) the signal has lowered

D. (1) all have partially opened
(2) the signal has increased

Answer: B

Answer Explanation

Refer to the DEHC Simplified Logic, shown below.

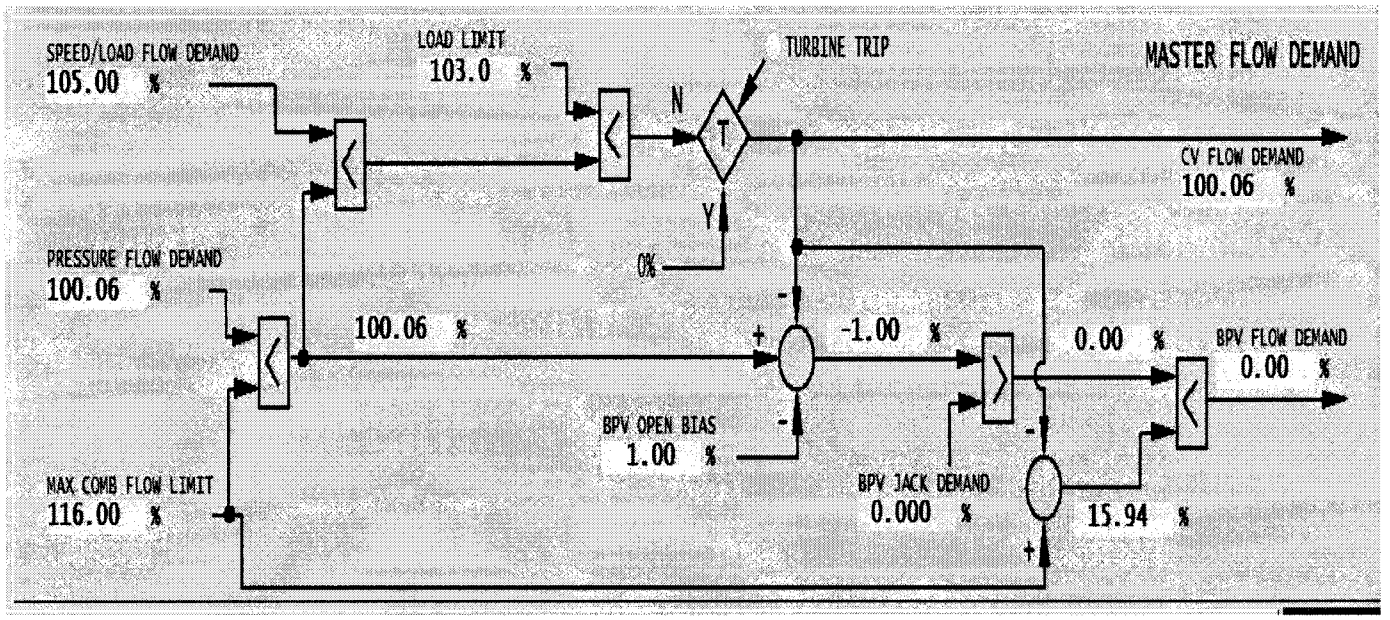
Initially, a 0% signal appears at the Bypass Control Unit HVG (>); it's the larger of the two input signals appearing there (the other signal being a -1% signal coming from the Bypass Control Unit summer), and so passes passes through the HVG to the LVG, where it is the smaller of two signals (the other being the ~16% signal from the MCFL summer). Thus, the 0% signal passes through the LVG to become a 0% BPV Flow Demand signal, keeping the BPVs closed.

When the operator presses the FULL OPEN button, the BPVs begin to ramp open, **in sequence.** Only when one BPV is fully open does the next BPV (in the sequence) begin to ramp open. If the operator does not press the HOLD button, all 9 BPVs will eventually ramp, in sequence, to 100% open.

Pressing the HOLD button simply pauses (stops) the ramping-up of the BPV Jack Demand signal. Because the operator has pressed HOLD just 10 seconds after pressing FULL OPEN, the result will be that one or more have opened, but not all of them.

Whatever the actual BPV Jack Demand signal value might be when the HOLD button is pressed, that signal is certainly larger than the prior 0% signal, discussed above. To make our point, suppose that signal is paused (HOLD) when it is 5%. This 5% signal is processed in exactly the same manner as earlier described for the 0% signal; except, now, a 5% BPV Flow Demand signal results in one or more of the BPVs partially opening, in sequence, enough to pass 5% steam flow to the main condenser.

The BPV opening causes a drop in reactor pressure and PAM pressure. The drop in PAM pressure results in a smaller output signal from the Pressure Control Unit, which still passes through both LVGs to ultimately become a new smaller CV Flow Demand (% open) signal, causing the TCVs to close enough to compensate for the steam flow being diverted to the main condenser via the open BPV.



'B' is correct: (1) one or more have opened, but not all of them; (2) the signal has lowered. Correct for the reasons described above.

'A' is wrong: (1) one or more have opened, but not all of them; (2) the signal has increased. Plausible to the examinee who does not fully comprehend how the Pressure Regulator functions; i.e., that when something occurs that results in bypassing some steam flow to the main condenser, the accompanying PAM pressure drop necessarily cause the TCVs to throttle closed enough to compensate for the steam loss and at the same time restore PAM pressure to the Pressure Set setpoint.

'C' is wrong: (1) all have partially opened; (2) the signal has lowered. Plausible to the examinee who still somewhat unfamiliar with the Unit 1 Digital EHC modification (installed during the Spring, 2014 refueling outage). That examinee incorrectly believes that the fundamental manner in which the various turbine valves, controlled by EHC, behave is different than before the modification. This is wrong thinking...the BPVs are still designed to open in sequence; they do not, under any circumstances, open simultaneously.

'D' is wrong: (1) all have partially opened; (2) the signal has increased. Plausible for reasons similar to that for choices 'A', 'C'.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 73 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	2.50
System ID:	1104005
User-Defined ID:	REV 01, 12/15/14
Cross Reference Number:	LGSOPS0031B.4
Topic:	Unit 1 DEHC - Predict BPV/TCV response to operating BPV Jack
Num Field 1:	3.8
Num Field 2:	3.9
Text Field:	241000 K1.06
Comments:	Level RO Tier 2 Group 2 KA # and Rating 241000 K1.06 (3.8/3.9) KA Statement 241000 Reactor/Turbine Pressure Regulator Knowledge of the physical connections and/or cause-effect relationships between REACTOR/TURBINE PRESSURE REGULATING SYSTEM and the following: K1.06 Bypass valves References DEHC Simplified Logic diagram Examinee References None Learning Objective LGSOPS0031B.4 Question source Bank 556763* Question history None Cognitive level Higher 10 CFR 55 41.5 Comments *Major revision to address NRC Rev.00 comments.

74 **ID: 1104006** **Points: 1.00**

An important plant system has become unavailable (unable to provide its design function).

WHICH ONE of the following must be controlled as "Protected Equipment", per OP-LG-108-117-1000 (Limerick Protected Equipment Program)?

The redundant train/system to be "protected", if lost, could result in...

A. any unplanned main generator load reduction

B. a PARAGON risk color of YELLOW

C. a main generator load reduction of >/= 15 MWe

D. a Tech Spec LCO Action requiring the plant to be in HOT SHUTDOWN within 12 hours

Answer: D

Answer Explanation

Refer to OP-LG-108-117-1000, Section 4.1.1.

'D' is correct: a Tech Spec LCO Action requiring the plant to be in HOT SHUTDOWN within 12 hours. Correct per Section 4.1.1.2.

'A' is wrong: any unplanned main generator laod reduction. Per Section 4.1.1.3, the threshold is > 20 MWe reduction in generator capacity. Plausible to the examinee who recalls something about an unplanned load reduction but doesn't recall there being a threshold that reduction.

'B' is wrong: a PARAGON risk color of YELLOW. Per Section 4.1.1.1, only if it would result in a RED risk color. Plausible to the examinee who recognizes the importance/"visibility" of on-line risk assessment and is therefore distracted by the idea of any increased risk. NOTE - "PARAGON" is the name of the computer program used at LGS to perform risk assessments.

'C' is wrong: a main generator load reduction of >/= 15 MWe. Per Section 4.1.1.3, the threshold is > 20 MWe reduction in generator capacity. Plausible to the examinee who recognizes a 15 MWe loss as being greater than 1% of the 100% power real load of approximately 1200 MWe.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 74 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	3.00	
System ID:	1104006	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	NONE	
Topic:	Recall Protected Equipment process requirements	
Num Field 1:	3.9	
Num Field 2:	4.3	
Text Field:	2.2.14	
Comments:	Level	RO
	Tier	3
	Group	N/A
	KA # and Rating	2.2.14 (3.9/4.3)
	KA Statement	2.2.14 Knowledge of the process for controlling equipment configuration or status.
	References	OP-LG-108-117-1000, Rev.5
	Examinee References	None
	Learning Objective	No specified objective
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.10
	Comments	None

75 **ID: 1104526** **Points: 1.00**

Plant conditions:

- Unit 1 is in OPCON 1
- Unit 2 is in OPCON 2 with GP-2, STARTUP in progress

A valid OBE EXCEEDED alarm is received and is confirmed by the U.S. Geological Survey.

WHICH ONE of the following is a required operator action?

A. Manually scram both reactors

B. Perform GP-4 Rapid Plant Shutdown on Unit 1
Manually scram Unit 2 reactor

C. Perform GP-4 Rapid Plant Shutdown on both Units

D. Perform GP-3 Normal Plant Shutdown on both Units

Answer: A

Answer Explanation

Per SE-5 (Earthquake), Steps 4.2 and 4.5. The "valid" comment in the stem translates to "evidence of the seismic event" (mentioned in Step 4.5). Though Step 4.5 is not an "Immediate Operator Action" initial license candidates are expected to recall manual scram/GP-4 Rapid Shutdown actions, no matter where they exist in procedure.

'A' is correct: Manually scram both reactors. Correct for the reasons described above.

'B' is wrong: Perform GP-4 Rapid Plant Shutdown on Unit 1; Manually scram Unit 2 reactor. Very plausible to the examinee who considers that, whenever practical, LGS procedures direct us to perform a GP-4 Rapid Plant Shutdown (i.e., running Recirc Pumps to minimum speed and transferring house loads to the startup buses before scramming the reactor) in order to minimize the plant transient. The fact that Unit 2 is only in OPCON 2 during a startup, the idea of simply scramming the reactor (i.e., without concern for the severity of a plant transient) makes this answer choice all the more distracting.

'C' is wrong: Perform GP-4 Rapid Plant Shutdown on both Units. Plausible for reasons similar to that for choice 'B'; except, here, the examinee considers it more prudent to minimize the plant transient for both Units. This examinee has neglected to consider that with Unit 2 in OPCON 2, its house loads are still on the startup buses and its Total Core Flow is well below 60% (i.e., the target flow for the ruduction in Recirc directed by GP-4).

'D' is wrong: Perform GP-3 Normal Plant Shutdown on both Units. Plausible to the examinee who does not recognize the significance of an OBE earthquake and/or confuses it with an SSE earthquake. As such, the examinee is compelled to believe that an orderly plant shutdown of both Units is sufficient to mitigate the effects of the OBE.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 75 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.00	
System ID:	1104526	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	LLOT1563.02	
Topic:	SE-5 - Recall Requirements for Manual Scram	
Num Field 1:	4.0	
Num Field 2:	4.2	
Text Field:	2.4.11	
Comments:	Level	RO
	Tier	3
	Group	N/A
	KA # and Rating	2.4.11 (4.0/4.2)
	KA Statement	2.4.11 Knowledge of abnormal condition procedures.
	References	SE-5, Rev.35
	Examinee References	None
	Learning Objective	LLOT1563.02 (in lesson plan LGSOPS2000)
	Question source	Bank 558648
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.10
	Comments	None

76 **ID: 1102804** **Points: 1.00**

SRO

Unit 1, GP-2 Normal Plant Startup is in progress, with the following:

- Control rod withdrawals are in progress
- Reactor power has reached 15%

Per Tech Spec 3.6.6.3.a, drywell/suppression chamber oxygen concentration must be < 4% by volume within 24 hours after raising power above 15%.

WHICH ONE of the following is correct regarding documentation of the stated TS requirement for the given plant conditions?

A. Document the requirement in the LCO Log

B. Initiate a Short Duration Time Clock (SDTC) Log entry

C. Documentation is NOT required

D. Initiate a Potential LCO Action Request (LCOAR)

Answer: A

Answer Explanation

Per GP-2, Step 3.5.5.5, [the CRS] must initiate an LCO Log entry to ensure that the applicable Surveillance Test is performed to verify <4% oxygen within 24 hours after raising power above 15% (i.e., the statement of LCO 3.6.6.3.a).

'A' is correct: Document the requirement in the LCO Log. Correct for the reason described above.

'B' is wrong: Initiate a Short Duration Time Clock (SDTC) Log entry. This log is described in OP-AA-108-104 (Technical Specfication Compliance), Section 2.1. The SDTC is used for tracking components made inoperable for testing when an LCO Action entry is not required. Because the SDTC "may also apply to short term LCOs...", it is especially plausible to the examinee who cannot recall the GP-2 requirement to use the LCO Log.

'C' is wrong: Documentation is NOT required. Plausible to the examinee who mistakenly believes that LCO log entries apply only when an LCO is actually entered.

'D' is wrong: Initiate a Potential LCO Action Request (LCOAR). The LCOAR terms refers simply to a type of Action Request generated using the computerized AR system. An LCOAR would be generated to track a component that, in itself, is known to be inoperable, but by itself, that inoperability doesn't yet require an LCO Action entry. Plausibility speaks for itself.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 76 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.50	
System ID:	1102804	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:		
Topic:	(SRO) Recall how to track an LCO during a normal plant startup	
Num Field 1:	3.1	
Num Field 2:	4.6	
Text Field:	2.2.23	
Comments:	Level	SRO
	Tier	3
	Group	N/A
	KA # and Rating	2.2.23 (3.1/4.6)
	KA Statement	2.2.23 Ability to track Technical Specification limiting conditions for operations.
	References	GP-2, Rev.153 OP-AA-108-104, Rev.1
	Examinee References	None
	Learning Objective	No specified objective
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	41.10, 43.2, 43.5
	Comments	Editorial changes (primarily to answer choices) to address NRC Rev.0 coments.

77 **ID: 1118645** **Points: 1.00**

SRO

Unit 1 is in OPCON 4 cooling down for a refueling outage, with the following:

- 'A' RHR is in Shutdown Cooling (SDC)
- Reactor Coolant System (RCS) temperature is 170°F

'A' RHR Pump trips on thermal overload and cannot be re-started.

At 1200:

- RCS temperature reaches 200°F, rising

At 1210:

- 'B' RHR is operating in SDC
- RCS temperature is 210°F, lowering

At 1245:

- RCS temperature is 205°F, lowering

WHICH ONE of the following identifies the **LATEST** time by which an ALERT Emergency Action Level must be declared based on this event, if at all?

A. NO declaration is required

B. 1215

C. 1220

D. 1300

Answer: A

Answer Explanation

Refer to EP-AA-1008, Cold Matrix, "Heat Sink", EAL: CA5. Threshold #1 applies for the context of this question. To "meet" that Threshold, there must have been an UNPLANNED loss of decay heat removal that results in RCS temperature exceeding 200°F for the applicable "Duration" identified in Table C2. With the plant in OPCON 4, it is clear that the RCS is "intact"; therefore, regardless of whether or not "Containment Closure" is still established, the applicable "Duration" is 60 minutes. However, the (*) Note at the bottom of that Table tells us that Threshold #1 is NOT applicable if..."within this time frame (i.e., 60 minutes, in this case)...an RCS heat removal system is in operation and RCS temperature is being reduced." Stem conditions indicate that the 60-minute clock should have started at 1200 hours (i.e., the time at which RCS temperature exceeded 200°F); however, just 10 minutes later (at 1210 hours), 'B' RHR SDC is operating and RCS temperature is lowering as a result. Therefore, NO declaration is required.

'A' is correct: NO declaration is required. Correct for the reasons described above.

'B' is wrong: 1215. As a closed-reference question, this choice is quite plausible to the examinee who only recognizes that an OPCON change (to OPCON 3) has occurred at 1200 hours, and recalls that generally (except where EALs state otherwise) the Emergency Director has 15 minutes, thereafter, to declare the related EAL. That examinee mistakenly believes that an unplanned OPCON change, by itself, is an EAL; it is not.

'C' is wrong: 1220. As a closed-reference question, this choice is plausible to the examine who vaguely recalls there being a "20 minute" duration included on Table C2 of EAL CA5, but who does not recall exactly when that 20 minute durations applies (per Table C2, it applies when the RCS is NOT intact but Containment Closure is established).

'D' is wrong: 1300. Plausible, again, to the examinee who fails to recall that the (*) Note of Table C2 excluded Threshold #1 as of 1210. However, that examinee also vaguely recalls something about a 60-minute duration on that Table. Recognizing that the OPCON change problem still exists at 1245 (i.e., still above 200°F), the examinee mistakenly believes that the OPCON change, itself, warrants an EAL declaration within the usual 15 minutes thereafter (i.e., 1245 + 15 = 1300).

NOTE - This question and its answer choices has intentionally ignored Threshold #2 of EAL CA5. That Threshold would require the E.D. to declare within the usual 15 minutes of after experiencing a 10 psig RPV pressure rise. Even with RCS temperature peaking at 210°F (at 1210 hours), there is no way for RPV pressurization of any amount to have occurred.

Question 77 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	2.50	
System ID:	1118645	
User-Defined ID:	REV.02, 01/06/15	
Cross Reference Number:	NONE	
Topic:	(SRO) Recall EAL declaration time requirement for loss of DHR capability	
Num Field 1:	2.9	
Num Field 2:	4.6	
Text Field:	295021 2.4.41	
Comments:	Level	SRO
	Tier	1
	Group	1
	KA # and Rating	295021 2.4.41 (2.9/4.6)
	KA Statement	295021 Loss of Shutdown Cooling 2.4.41 Knowledge of the emergency action level thresholds and classifications.
	References	EP-AA-1008, Rev.27 Cold Matrix
	Examinee References	None
	Learning Objective	No specified objective
	Question source	New
	Question history	None
	Cognitive level	Lower

	10 CFR 55 Comments	43.5 Replacement question to address NRC Rev.00 comments. Minor change (stem condition) to improve plausibility of choice 'D', to address NRC Rev.01 comments.

78 **ID: 1102809** **Points: 1.00**

SRO

Unit 2 is operating at 100% power when the following occurs:

- Complete sustained loss of RECW

Per ON-113 (Loss of RECW), operators:

- Perform a GP-4 Rapid Plant Shutdown
- Trip both Recirc Pumps

After the successful scram, operators commence a controlled plant cooldown.

WHICH ONE of the following identifies the NRC ENS Notification required for this event?

A. 1 hour per SAF for Declaration of Emergency Class

B. 4 hours per SAF for Plant Shutdown Required by Tech Specs

C. 4 hours per SAF for Reactor Protection System Actuation

D. 8 hours per SAF for Degraded or Unanalyzed Condition

Answer: C

Answer Explanation

Refer to the LGS Emergency Action Level (EAL) "Hot Matrix". Nothing in the stem conditions indicates an EAL threshold has been met; thus, no Emergency Classification is required. Refer to Exelon Reportability Manual, LS-AA-1110, specifically:

- SAF 1.1 (Declaration of Emergency Class), a 1-hour report, which does not apply for this event.
- SAF 1.2 (Plant Shutdown Required by Tech Specs), a 4-hour report. This SAF does not apply for this event because the reason operators having inserted the manual scram (per ON-113, Loss of RECW) is not tied to any Tech Spec action. Rather, the reason is based on a concern for potential Thermal Hydraulic Instabilities (THI) if the reactor were to continue operating with no operating Recirc Pumps (refer to ON-113 basis "Discussion" of Section 4.0).
- SAF 1.4 (Degraded or Unanalyzed Condition), an 8-hour report. This SAF does not apply for this event because those reporting critieria are concerned only with a "serious degradation" of the plant's principle safety barriers, or the plant operating in some type of "unanalyzed" condition. Nothing in the stem indicates any of those criteria exist.
- SAF 1.6 (RPS Actuation), a 4-hour report. A thorough review reveals that any RPS actuation (so long as it is neither an "invalid" one, nor one part of "preplanned testing") is included in those reportting criteria. The manual scram that occurred for this Loss of RECW event warrants an NRC ENS notification per SAF 1.6

'C' is correct: 4 hours per SAF for Reactor Protection System Actuation. Correct for the reasons described above.

'A' is wrong: 1 hour per SAF for Declaration of Emergency Class. Plausible to the examinee who mistakenly believes that the event meets the threshold for EAL "MU7" (Inability to reach required shutdown within Tech Spec limits).

'B' is wrong: 4 hours per SAF for Plant Shutdown Required by Tech Specs. Plausible to the examinee who inappropriately ties the shutdown required by ON-113 with the shutdown required by Tech Spec 3.4.1.1, specifically Action 'b' which requires the plant be in HOT SHUTDOWN within 12 hours of having no operating Recirc Pumps.

'D' is wrong: 8 hours per SAF for Degraded or Unanalyzed Condition. Plausible to the examinee who does recall the ON-113 bases regarding inserting a manual scram to preclude the possibility of THI and therefore concludes that the need to avoid THI (an unanalyzed condition) fits with the reporting criteria of SAF 1.4.

Question 78 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	5	
Difficulty:	3.00	
System ID:	1102809	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LGSOPS1550	
Topic:	(SRO) Determine NRC Reportability for Loss of RECW	
Num Field 1:	2.7	
Num Field 2:	4.1	
Text Field:	295018 2.4.30	
Comments:	Level	SRO
	Tier	1
	Group	1
	KA # and Rating	295018 2.4.30 (2.7/4.1)
	KA Statement	295018 Partial or Total Loss of CCW 2.4.30 Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.
	References	ON-113, Rev.23 U/1 Tech Spec 3.4.1.1, latest LS-AA-1110, Rev.21 EP-AA-1008, Rev.26
	Examinee References	None
	Learning Objective	LGSOPS1550, no specified objective

	Question source	New
	Question history	None
	Cognitive level	Higher
	10 CFR 55	43.5
	Comments	Editorial change (answer choices) to address NRC Rev.00 comments.

79 **ID: 1102824** **Points: 1.00**

SRO

Unit 1 is operating at 100% power.

Operators determine that Suppression Pool water level is 21 feet, down slow.

WHICH ONE of the following identifies the plant shutdown requirement, directed by either Tech Specs or T-102, if Suppression Pool level cannot be restored?

A. Be in STARTUP within 7 hours

B. Perform a T-102 shutdown if the level cannot be maintained above 18 feet

C. Be in HOT SHUTDOWN within 13 hours

D. Perform a T-102 shutdown if the level cannot be maintained above 13.5 feet

Answer: C

Answer Explanation

The 21 foot suppression pool water level requires an entry into Tech Spec LCO 3.6.2.1, as well as antry into T-102 (Primary Containment Control). LCO 3.6.2.1 Action 'a' requires that level be restored within 1 hour, otherwise place the plant in HOT SHUTDOWN within the next 12 hours (i.e., a total of 13 hours from the time of discovering the out-of-spec level).

The Suppression Pool Level Leg (SP/L) of T-102 requires that HPCI be secured (if running) if level cannot be maintained above 18 feet (Step SP/L-4) and requires operators to either swap ECCS pump suctions over to the CST, or secure those pumps, if level cannot be maintained above 13.5 feet (Step SP/L-5). When level cannot be maintained above 12 feet, a rapid plant shutdown (known as a "T-102 shutdown") is required (Steps SP/L-7 and 8).

'C' is correct: Be in HOT SHUTDOWN within 13 hours. Correct for the reasons described above.

'A' is wrong: Be in STARTUP within 7 hours. This alludes to the plant shutdown required by LCO 3.0.3 (i.e., 1 hour to initiate action to shut down the plant plus another 6 hours to be in STARTUP (OPCON 2), for a total of 7 hours). Plausible to the examinee who recognizes that the suggested T-102 actions are wrong, but who does not recall the LCO 3.6.2.1 Actions, or who mistaknely believes that LCO 3.0.3 applies for this situation.

'B' is wrong: Perform a T-102 shutdown if the level cannot be maintained above 18 feet. Plausible to the examinee who fails to recall the LCO Actions, is distracted by the SP/L leg of T-102, and fails to recall that a T-102 shutdown isn't required until level gets below 12 feet.

'D' is wrong: Perform a T-102 shutdown if the level cannot be maintained above 13.5 feet. Plausible for reasons identical to that for choice 'C'.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 79 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.50	
System ID:	1102824	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LLOT0060A.IL10	
Topic:	(SRO) Recall plant shutdown requirements TS 3.6.2.1 for low supp pool level	
Num Field 1:	3.4	
Num Field 2:	4.7	
Text Field:	295030 2.2.40	
Comments:	Level	SRO
	Tier	1
	Group	1
	KA # and Rating	295030 2.2.40 (3.4/4.7)
	KA Statement	295030 Low Suppression Pool Water Level 2.2.40 Ability to apply Technical Specifications for a system.
	References	U/1 Tech Spec 3.6.2.1, latest T-102, Rev.24
	Examinee References	None
	Learning Objective	LLOT0060A.IL10
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	43.2
	Comments	Editorial change to stem to address NRC Rev.00 coments.

80 **ID: 1102826** **Points: 1.00**

SRO

Plant conditions:

- Unit 1 is in OPCON 5, with CORE ALTERATIONs in progress
- Unit 2 is operating at 100% power

Due to heavy offsite grid loading, operators are executing E-5 (Grid Emergency).

Operators determine that actual voltage on the 500 KV System is 490 KV.

WHICH ONE of the following identifies the Tech Spec required action, if any, for each Unit?

A. Unit 1: No Tech Spec required action
Unit 2: Restore the 500 KV system to OPERABLE within 72 hours

B. Unit 1: No Tech Spec required action
Unit 2: Restore the 500 KV system to OPERABLE within 24 hours

C. Unit 1: Immediately suspend CORE ALTERATIONs
Unit 2: Restore the 500 KV system to OPERABLE within 72 hours

D. Unit 1: Immediately suspend CORE ALTERATIONs
Unit 2: Restore the 500 KV system to OPERABLE within 24 hours

Answer: A

Answer Explanation

Per E-5, Section 3.13.1 and its Table, the minimum allowable voltage on the 500 KV system is 498 KV. With only 490 KV actual voltage, Step 3.13.1 directs operators declare the [affected] "offsite source" inoperable and take the Action of Tech Spec 3.8.1. An inoperable 500 KV system takes away only one Tech Spec offsite source from each of the LGS Units.

Per Unit 1 Tech Spec 3.8.1.2 (AC Sources - Shutdown), only one of the two offsite sources is required to be OPERABLE. Since the 230 KV system is still OPERABLE, there is no LCO 3.8.1.2 entry required for Unit 1.

Per Unit 2 Tech Spec 3.8.1.1 (AC Sources - Operating), two offsite sources are required to be OPERABLE. With the 500 KV system inoperable, LCO 3.8.1.1 entry is required and ACTION 'f' applies...allowing 72 hours to restore the 500 KV system to OPERABLE status.

'A' is correct: Unit 1: No Tech Spec required action; Unit 2: Restore the 500 KV system to OPERABLE within 72 hours. Correct for the reasons described above.

'B' is wrong: Unit 1: No Tech Spec required action; Unit 2: Restore the 500 KV system to OPERABLE within 24 hours. Plausible to the examinee who cannot recall the Allowed Outage Time (AOT) for a single inop offsite source and so applies the two inop offsite sources AOT of ACTION 'g', which is 24 hours.

'C' is wrong: Unit 1: Immediately suspend CORE ALTERATIONs; Unit 2: Restore the 500 KV system to OPERABLE within 72 hours. Part (1) would be correct if it were the 230 KV System that is inoperable. Per E-5, such would require that BOTH offsite sources (to both the 10 Startup Bus and 20 Startup Bus) be declared inoperable. In that case, with no (zero) offsite sources operable, Unit 1 Tech Spec Action 3.8.1.2.a would apply, requiring the immediate suspension of Core Alterations. Plausible for these reasons.

'D' is wrong: Unit 1: Immediately suspend CORE ALTERATIONs; Unit 2: Restore the 500 KV system to OPERABLE within 24 hours. This choice would be correct if it were the 230 KV System that is inoperable. Per E-5, such would require that BOTH offsite sources (to both the 10 Startup Bus and 20 Startup Bus) be declared inoperable. In that case, with no (zero) offsite sources operable, Unit 2 Tech Spec Action 3.8.1.1.g would apply, requiring the restoration of the 230 KV system to OPERABLE within 24 hours. Similarly, Unit 1 Tech Spec Action 3.8.1.2.a would apply, requiring the immediate suspension of Core Alterations. Plausible for these reasons.

Question 80 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	2.50	
System ID:	1102826	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LLOT0035.11B	
Topic:	(SRO) Evaluate impact of exceeding the 500 KV system minimum voltage limit	
Num Field 1:	3.2	
Num Field 2:	3.8	
Text Field:	700000 AA2.05	
Comments:	Level	SRO
	Tier	1
	Group	1
	KA # and Rating	700000 AA2.05 (3.2/3.8)
	KA Statement	700000 Generator Voltage and Electric Grid Disturbances Ability to determine and/or interpret the following as | they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: AA2.05 Operational status of offsite circuit
	References	E-5, Rev.21 U/1 Tech Spec 3.8.1.2, latest U/2 Tech Spec 3.8.1.1, latest
	Examinee References	E-5 (***EXCLUDING*** the Attachments)
	Learning Objective	LLOT0035.11B
	Question source	New
	Question history	None
	Cognitive level	Higher
	10 CFR 55	43.2, 43.5

	Comments	Minor revision (one distractor, one Part) to address NRC Rev.00 comments.

81 **ID: 1104650** **Points: 1.00**

SRO

Unit 2 is operating at 100% power.

The CRS reviews the latest "Daily Jet Pump Operability" surveillance (ST-6-043-320-2) which reveals the following Conditions:

1. Indicated recirc loop flow differs by 11% from the established pump speed-loop flow characteristics
2. Indicated total core flow differs by 6% from the established total core flow derived from recirc loop flow measurements
3. Indicated diffuser-to-lower plenum d/p of a single jet pump (#3) differs from the established patterns by 14%
4. Indicated diffuser-to-lower plenum d/p for all other jet pumps differ from the established patterns by 8%

WHICH ONE of the following identifies (from the above list) the test data that requires the plant must be in HOT SHUTDOWN within 12 hours, per Tech Spec 3.4.1.2 (Jet Pumps)?

A. Condition 3, alone

B. Conditions 3 AND 4, combined

C. Conditions 1 AND 3, combined

D. Conditions 1, 2, AND 3, combined

Answer: ~~C~~ RO 3/10/15 Accept C or D

Answer Explanation

Refer to Tech Spec 3.4.1.2, SR 4.4.1.2.a, which shows only three distinct "conditions" (as they are called in the SR language) that are used to determine the OPERABILITY of the jet pumps (i.e., to determine the existence of any failed jet pump). Those "conditions" are the first three conditions given in the question stem. The 4th "condition" in the stem doesn't exist in the SR as a stand-alone item. Per the SR, if any two (never just one) of the three conditions exist, together, the presence of a failed jet pump is presumed and the ACTION of Tech Spec 3.4.1.2 (12-hour HOT SHUTDOWN) must be taken.

'C' is correct: Conditions 1 AND 3, combined. Both Condition1 and Condition 3 exceed the 10% limit of the SR.

'A' is wrong: Condition 3, alone. Plausible to the examinee who recognizes that Condition 1 exceeds the associated 10% limit, but who forgets that it takes at least two of the three Conditions to be UNSAT before having to take the plant shutdown ACTION.

'B' is wrong: Conditions 3 AND 4, combined. Plausible to the examinee who "seems to recall" something about "at least two" Conditions needing to be UNSAT in order to determine a failed jet pump, but who doesn't recall exactly which two Conditions. That examinee is distracted by the fact that a single jet pump (#3) is behaving significantly different from the other 19 jet pumps, in that its d/p differs by more than 5%. To the uncertain examinee, the suggestion of 5% is plausible considering the 5% recirc loop flow mismatch limit (of Tech Spec 3.4.1.3) that exists right now, with the plant operating at 100% power (i.e., well above 70% of rated Core Flow).

'D' is wrong: Conditions 1, 2, AND 3, combined. Plausible to the examinee who forgets that it takes only two UNSAT Conditions in order to have to take the plant shutdown ACTION. That examinee is distracted by the fact that Condition 2 is greater than 5%, confusing that with the 5% recirc loop flow mismatch limit (of Tech Spec 3.4.1.3) that exists right now, with the plant operating at 100% power (i.e., well above 70% of rated Core Flow).

RLP 3/10/15 Accept C or D

Question 81 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	3.00	
System ID:	1104650	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	NONE	
Topic:	(SRO) Recall Tech Spec 3.4.1.2 SR requirements for indications of a failed jet pump	
Num Field 1:	3.7	
Num Field 2:	4.1	
Text Field:	295001 2.2.12	
Comments:	Level	SRO
	Tier	1
	Group	1
	KA # and Rating	295001 2.2.12 (3.7/4.1)
	KA Statement	295001 Partial or Complete Loss of Forced Core Flow Circulation 2.2.12 Knowledge of surveillance procedures.
	References	U/2 Tech Spec 3.4.1.2 (latest) U/2 Tech Spec 3.4.1.3 (latest)
	Examinee References	None
	Learning Objective	No specified objective
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	43.2
	Comments	None

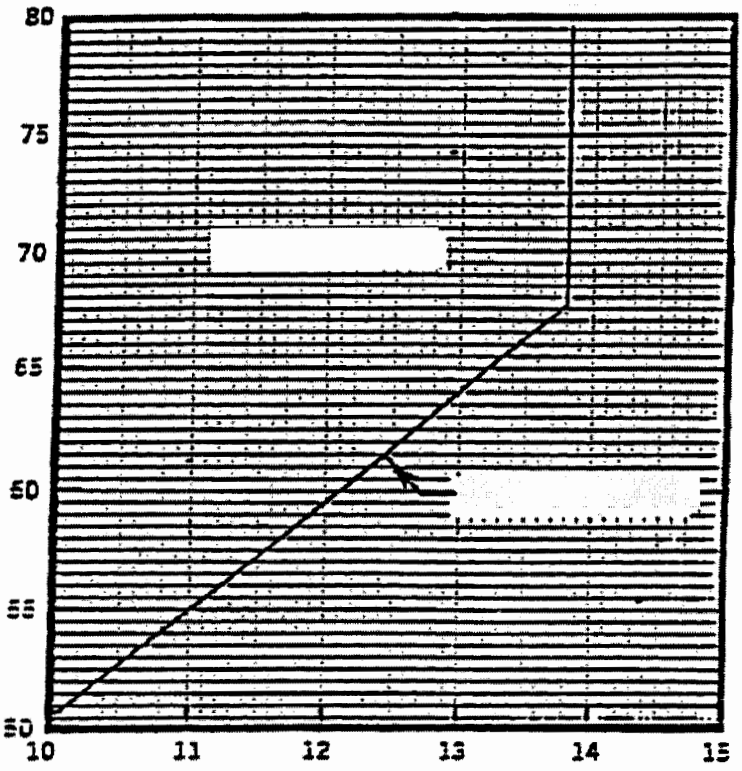
82 **ID: 1102864** **Points: 1.00**

SRO

Unit 1 is in OPCON 1, with the following:

- SLC Tank level is 4000 gallons
- SLC Tank temperature is 78°F
- SLC Tank Sodium Pentaborate Concentration is 14.5% by weight

WHICH ONE of the following identifies the Tech Spec required action, if any?



A. Adjust SLC Tank Level within 4 hours

B. Adjust SLC Tank Level within 8 hours

C. Adjust SLC Tank Concentration within 4 hours

D. Adjust SLC Tank Concentration within 8 hours

Answer: D

Answer Explanation

Refer to Tech spec 3.1.5:

- SR 4.1.5.a.2 requires at least 3160 gallons in the tank
- Per Figure 3.1.5-1, the 78°F tank temperature is below the 80°F limit
- Per Figure 3.1.5-1, the 14.5% concentration EXCEEDS the 13.8% limit for a tank temperature of 78°F

'D' is correct: Adjust SLC Tank Concentration within 8 hours. This is ACTION 'b'.

'A' is wrong: Adjust SLC Tank Level within 4 hours. Plausible to the examinee who cannot recall the 3160 gallon requirement of SR 4.1.5.a.2.

'B' is wrong: Adjust SLC Tank Level within 8 hours. Plausible to the examinee who recognizes that the worst-case (shortest AOT) associated with TS 3.1.5 is 8 hours, but who cannot recall the 3160 gallon requirement of SR 4.1.5.a.2.

'C' is wrong: Adjust SLC Tank Concentration within 4 hours. Plausible to the examinee who correctly interprets the un-labeled SLC TS Figure, but who cannot recall that the shortest AOT is 8 hours, not 4 hours.

Question 82 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	1102864
User-Defined ID:	REV 01, 12/15/14
Cross Reference Number:	LLOT0048.13
Topic:	(SRO) Determine TS Action for inoperable SLC
Num Field 1:	3.4
Num Field 2:	4.7
Text Field:	211000 2.2.40
Comments:	Level SRO Tier 2 Group 1 KA # and Rating 211000 2.2.40 (3.4/4.7) KA Statement 211000 SLC 2.2.40 Ability to apply Technical Specifications for a system. References U/1 Tech Spec 3.1.5, latest Examinee References None Learning Objective LLOT0048.13 Question source New* (was previously Bank) Question history None Cognitive level Higher 10 CFR 55 43.2 Comments *Major revision to address NRC Rev.00 comments.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

83 **ID: 1118706** **Points: 1.00**

SRO

Unit 1 is operating at 100% power when the following alarms are received:

- 1A INSTRUMENT AIR LO PRESSURE
- 1B INSTRUMENT AIR LO PRESSURE

Operators enter ON-119, Loss of Instrument Air, and start non-running air compressors.

WHICH ONE of the following describes the **NEXT** action required by ON-119?

A. IF the SCRAM VALVE PILOT AIR HEADER LO PRESSURE alarm is received, THEN reduce reactor power to less than 44%

B. IF both Instrument Air Header Pressure Indicators (PI-15-120A and B) are reading less than 85 psig, THEN reduce reactor power to less than 44%

C. IF the SCRAM VALVE PILOT AIR HEADER LO PRESSURE alarm is received, THEN manually scram the reactor

D. IF both Instrument Air Header Pressure Indicators (PI-15-120A and B) are reading less than 85 psig AND the SCRAM VALVE PILOT AIR HEADER LO PRESSURE alarm is received, THEN manually scram the reactor

Answer: B

Answer Explanation

Refer ON-119 (Loss of Instrument Air), Section 2.0. Stem indicates that Step 2.1 is already completed. The "NEXT" (decisive) action is defined by Step 2.3, where operators are directed to reduce power to less than 44% IF both air header pressure indicators are reading less than 85 psig. Although the alarm setpoint for the two alarms identified in the stem is 85 psig, operators must validate the alarms with the pressure indicators before taking any action to reduce power. Refer to the ON-119 Bases for Step 2.3, which shows that this action is performed in order to improve the operating margin for the feedwater control system to maintain adequate feedwater to the RPV in the event that the RFP min flow valves or Condensate Pump min flow valve begin to drift open due to low air header pressure.

'B' is correct: IF both Instrument Air Header Pressure Indicators (PI-15-120A and B) are reading less than 85 psig, THEN reduce reactor power to less than 44%. Correct for the reasons described above.

'A' is wrong: IF the SCRAM VALVE PILOT AIR HEADER LO PRESSURE alarm is received, THEN reduce reactor power to less than 44%. This alarm is the other entry condition for ON-119 (see Symptom 1.2). There is no correlation between a low scram air header pressure and the need to reduce power. Plausible to the Examinee who cannot recall the Bases for the power reduction of Step 2.3

'C' is wrong: IF the SCRAM VALVE PILOT AIR HEADER LO PRESSURE alarm is received, THEN manually scram the reactor. This alarm is the other entry condition for ON-119 (see Symptom 1.2). While Steps 2.3.2 and 2.3.3 consider the need to manually scram in the vent that rods beging to drift as a result of a low scram air header pressure, there is no need to scram the unit simply becuase of the receipt of this alarm (whose setpoint is 65 psig, per the Bases for Step 1.2). Plausible to the Examinee who mistakenly confuses the requirement to manually scram when rods beging to drift with the receipt of the scram air header pressure alarm that will preceed the point at which rod drifts are likely.

'D' is wrong: IF both Instrument Air Header Pressure Indicators (PI-15-120A and B) are reading less than 85 psig AND the SCRAM VALVE PILOT AIR HEADER LO PRESSURE alarm is received, THEN manually scram the reactor. Plausible to the Examinee who has a weak recollection of Section 2.0 actions in ON-119. That Examinee incorrectly believes that the direction to scram the reactor is premised on a very conservation approach; i.e., one that requires both the validation of the low air header pressure alarms AND the receipt of the scram air header low pressure alarm.

Question 83 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.00	
System ID:	1118706	
User-Defined ID:	REV 02, 01/06/15	
Cross Reference Number:	NONE	
Topic:	(SRO) ON-119 - Recall Subsequent Actions	
Num Field 1:	4.0	
Num Field 2:	4.2	
Text Field:	300000 2.4.11	
Comments:	Level	SRO
	Tier	2
	Group	1
	KA # and Rating	300000 2.4.11 (4.0/4.2)
	KA Statement	300000 Instrument Air 2.4.11 Knowledge of abnormal condition procedures.
	References	ON-119, Rev.27 ON-119 Bases, Rev.27
	Examinee References	None
	Learning Objective	No specified objective
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	43.5
	Comments	Replacement "new" question to address NRC Rev.00 comments.

84 **ID: 1102904** **Points: 1.00**

SRO

Unit 1 is operating at 100% power.

RWCU trips and isolates due to a Filter/Demin controller malfunction.

WHICH ONE of the following describes the Tech Spec or TRM required action, if any?

A. No Tech Spec or TRM required action

B. Direct Chemistry to sample/measure reactor coolant conductivity once every 4 hours

C. Direct Chemistry to sample/measure reactor coolant conductivity once every 24 hours

D. Verify both Reactor Water Sample Valves (HV-043-1F019(020)) are open within 2 hours

Answer: B

Answer Explanation

RWCU doesn't impact Tech Specs, directly or indirectly. However, its loss does impact TRM 3.4.4, Reactor Coolnat System Chemistry. Specifically, SR 4.4.4.1.c requires a continuous in-line conductivity measurement; if that is not available, then reactor coolant cunductivity must be sampled/measured every 4 hours, while in OPCON 1, 2, 3 (every 24 hours in OPCON 4, 5). Reactor coolant water conductivity is continuously measured at three points in the RWCU system...at the Outlet of the NRHX (also known as the Demin Inlet) and at the Oultet of both RWCU Filter/Demins...and is displayed on two MCR recorders (CR-1R601 and 1R603) at panel 10C602 (refer to alarm response card ARC-MCR-112, G4 to validate this).

Recirc Loop 'B' includes two, normally-open, inboard/outboard Reactor Water Sample Valves (HV-043-1F019 and 1F020) that provide a portion of that Loop's water to a Process Sampling Station (refer to P&ID M-0043, Sheet 1, coordinates G/2-4 to validate this). It is at that Sampling Station where personnel can take a grab sample of reactor coolant.

'B' is correct: Direct Chemistry to sample/measure reactor coolant conductivity once every 4 hours. Correct for the reasons described above.

'A' is wrong: No Tech Spec or TRM required action. Plausible to the inexperienced operator/examinee who fails to recognize the connection between an in-service RWCU system and the continuous in-line conductivity monitor.

'C' is wrong: Direct Chemistry to sample/measure reactor coolant conductivity once every 24 hours. This would be correct were the plant in OPCON 4 or 5 (as earlier discussed). Plausible to examinee who fails to recall the 4-hour requirement of TRM SR 4.4.4.1.c.1.

'D' is wrong: Verify both Reactor Water Sample Valves (HV-043-1F019(020)) are open within 2 hours. Very plausible to the examinee who believes that this sample line also includes a continuous in-line conductivity monitor.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 84 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.50	
System ID:	1102904	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LGSOPS0044.IL11	
Topic:	(SRO) RWCU Isolation - Determine Tech / TRM Action	
Num Field 1:	3.6	
Num Field 2:	4.5	
Text Field:	204000 2.2.38	
Comments:	Level	SRO
	Tier	2
	Group	2
	KA # and Rating	204000 2.2.38 (3.6/4.5)**
	KA Statement	204000 RWCU 2.2.38 Knowledge of conditions and limitations in the facility license
	References	U/1 TRM 3.4.4, latest P&ID M-0043, Sheet 1, Rev.51 ARC-MCR-112, G4, Rev.2
	Examinee References	None
	Learning Objective	LGSOPS0044.IL11
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	43.1, 43.2
	Comments	**Justification for KA match: There is no way to connect RWCU with the Facility Operating License (FOL); however, the facility Tech Specs are part of that FOL (in fact, just an Ammendment to the FOL). Therefore, a question framed as it would be were the selected KA 2.2.40, instead, is also an appropriate match with KA 2.2.38. Fixed typo in stem to address NRC Rev.00 comments.

85 **ID: 1102906** **Points: 1.00**

SRO

Unit 1 plant conditions:

- Fuel Shuffle Part 2 is in progress
- A grappled fuel bundle is hanging above its "move to" location in the core
- The FHD and Spotter perform concurrent verification of location and orientation
- The Unit RO is informed that the bundle is going into the core

With the bundle fully seated and still grappled, the spotter reports that the bundle location is not correct.

WHICH ONE of the following describes the FH-105 (Core Component Movement - Core Transfers) requirement for correcting this fuel bundle location problem?

A. May be immediately corrected with concurrent verification steps re-performed on the same CCTAS

B. May be immediately corrected provided concurrent verification on the same CCTAS is performed by a different set of individuals

C. Shall NOT be corrected until the FHD authorizes a pen-and-ink change to the CCTAS

D. Shall NOT be corrected until the concurrent verification failure is reported to Shift Management, RSS Management and Reactor Engineering

Answer: D

Answer Explanation

'D' is correct: Shall NOT be corrected until the concurrent verification failure is reported to Shift Management, RSS Management and Reactor Engineering. Taken directly from FH-105, Step 8.4.1.

'A' is wrong: May be immediately corrected with concurrent verification steps re-performed on the same CCTAS. This would be a violation of FH-105. Plausible to the examinee who believes the bundle should be placed in its correct location prior to suspending fuel moves.

'B' is wrong: May be immediately corrected provided concurrent verification on the same CCTAS is performed by a different set of individuals. Again, this violates FH-105. Plausbile to the examinee who believes that placing the bundle in the correct location is the correct action and that it's acceptable to re-perform the move with two different individuals.

'C' is wrong: Shall NOT be corrected until the FHD authorizes a pen-and-ink change to the CCTAS. This choice is taken from FH-105, Step 8.5, which states..."IF it becomes necessary to deviate from the Move Sheet, THEN the Fuel Handling Director shall halt core component moves prior to performing the step requiring deviation AND initiate a revision to the Move Sheet NF-AA-309". NF-AA-309 (SNM and Core Component Move Sheet Development) requires Reactor Engineering approval of move sheet changes. This choice is plausible to the examinee who believes the Fuel Handling Director can approve a pen and ink change to a move sheet without RE approval.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 85 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.50	
System ID:	1102906	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LLOT0760	
Topic:	(SRO) Recall FH-105 requirements for correcting a fuel bundle mis-located in the core	
Num Field 1:	2.5	
Num Field 2:	3.4	
Text Field:	2.1.42	
Comments:	Level	SRO
	Tier	3
	Group	N/A
	KA # and Rating	2.1.42 (2.5/3.4)
	KA Statement	2.1.42 Knowledge of new and spent fuel movement procedures.
	References	FH-105, Rev.47
	Examinee References	None
	Learning Objective	LLOT0760, no specified objective
	Question source	Bank 985947
	Question history	None
	Cognitive level	Lower
	10 CFR 55	43.7
	Comments	Re-classified as Lower Cognitive to address NRC Rev.00 comments.

86 **ID: 1104204** **Points: 1.00**

SRO

Unit 1 is operating at 100% power.

Operators have just completed ST-6-001-761-1, "Main Turbine Bypass Valve (BPV) Exercising".

The ST indicates that 3 BPVs will not open should a main turbine trip occur.

CRS declares the 3 BPVs inoperable.

WHICH ONE of the following describes the impact on continued plant operation?

A. May continue operating at 100% power but with a more limiting MCPR value

B. May continue operating at 100% power with the same MCPR limits

C. May continue operating only at power levels above 90% or below 25%; no Thermal limits apply

D. May continue operating only at power levels below 25%; no Thermal limits apply

Answer: A

Answer Explanation

Refer to U/1 TS 3.7.8 (Main Turbine Bypass System). The LCO statement points to the COLR to determine the minimum number of BPVs required to be OPERABLE. Refer to the U/1 COLR, Section 7.0, where we find that a minimum of 7 valves are required. Stem indicates that only 6 of the 9 BPVs are OPERABLE; therefore, an LCO entry is required. The sole ACTION (of TS 3.7.8) is...With the main turbine bypass system inoperable, restore the system to OPERABLE status within 1 hour or take the ACTION required by Specification 3.2.3.c. Refer to TS 3.2.3 (MCPR), where ACTION 'c' states...With the main turbine bypass system inoperable per Specification 3.7.8, operation may continue provided that, within 1 hour, MCPR is determined to be greater than or equal to the rated MCPR limit as a function of the average scram time (shown in the CORE OPERATING LIMITS REPORT) main turbine bypass valve inoperable curve, adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT.

'A' is correct: May continue operating at 100% power but with a more limiting MCPR value.
Correct for the reasons described above.

'B' is wrong: May continue operating at 100% power with the same MCPR limits. Plausible to the examinee who either does not recall that the COLR requires at least 7 BPVs to be OPERABLE, or who does, but does not recognize that a more limiting MCPR limit applies due to the inoperable turbine bypass system.

'C' is wrong: May continue operating only at power levels above 90% or below 25%; no Thermal limits apply. This choice "suggests" the idea of OT-102 (High Reactor Pressure), Attachment 3, which reminds operators that MCPR in unanalyzed at power levels between 25% and 90% with only a single functioning Pressure Regulator. Plausible to the examinee who confuses the two very different situations (inoperable Regulator versus inoperable turbine bypass system).

'D' is wrong: May continue operating only at power levels below 25%; no Thermal limits apply. Plausible to the examinee does nor recall that there is a specific MCPR LCO Action (3.2.3.c) that accounts for an inoperable turbine bypass system. This examinee recognizes that we are below the COLR minimum for operable BPVs and the fact that MCPR is somehow impacted, but thinks the only solution to reduce power to below 25% so that none of the Thermal Limits (MCPR, APLHGR, LHGR) apply.

Question 86 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	6	
Difficulty:	2.50	
System ID:	1104204	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:		
Topic:	(SRO) Determine Impact of Inop BPVs on Continued Plant Operation	
Num Field 1:	3.1	
Num Field 2:	3.1	
Text Field:	295005 AA2.03	
Comments:	Level	SRO
	Tier	1
	Group	1
	KA # and Rating	295005 AA2.03 (3.1/3.1)*
	KA Statement	295005 Main Turbine Generator Trip Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: AA2.03 Turbine valve position
	References	U/1 Tech Spec 2.1.2 (latest) U/1 Tech Spec 3.2.3 (latest) U/1 Tech Spec 3.7.8 (latest) U/1 COLR, Rev.11 OT-102, Rev.22 U/1 Tech Spec 3.7.8 BASES (latest)** U/1 UFSAR Section 15.1.2**
	Examinee References	None
	Learning Objective	No specified objective
	Question source	New
	Question history	None
	Cognitive level	Higher
	10 CFR 55	43.2

	Comments	*Justification for KA match: The "Basis" for Tech Spec 3.7.8 is to prevent the MCPR Safety Limit from being exceeded in the event of a feedwater controller failure concurrent with some degree of turbine bypass valve failure to respond. Refer to UFSAR Section 15.1.2 (Feedwater Controller Failure), specifically, Section 15.1.2.2.1, where the Sequence of Events includes the main turbine trip on high RPV water level. Then, refer to Section 15.1.2.3.3, where the the analysis "Results" describe the importance of the turbine bypass valves opening enough to limit peak pressure at the bottom of the RPV to 1218 psig. Based on this analysis, we regard this question as a suitable match for the sampled KA. **References added to support Rev.01. --- Editorial change (to this page only) to address NRC Rev.00 comments.

87 **ID: 1104205** **Points: 1.00**

SRO

Unit 1 is operating at 100% power.

Due to a degrading main condenser vacuum, operators insert a manual scram.

Current conditions:

- Mode Switch is in SHUTDOWN
- Two rods remain FULL OUT
- SRV 1E inadvertently opened and reclosed after fuses were pulled
- RPV level is +10", up slow
- Suppression Pool temperature is 112°F, up slow
- All "APRM DOWNSCALE" lights are lit

WHICH ONE of the following identifies the **WIDEST** allowable RPV level band?

A. -161" to +12.5", per T-101 (RPV Control)

B. -186" to +10", per T-117 (Level/Power Control)

C. +12.5" to +54", per T-101 (RPV Control)

D. -186" to +54", per T-117 (Level/Power Control)

Answer: D

Answer Explanation

On the scram, operators enter T-101 (RPV Control) due to RPV level (+12.5"). Upon discovery of "two rods remaining FULL OUT", they enter T-117 (Level/Power Control) and execute the LQ leg...with reactor power below 4% and RPV level not intentionally lowered, they arrive at Step LQ-16. The RPV level band at LQ-16 is -186" to +54".

'D' is correct: -186" to +54", per T-117 (Level/Power Control). Correct for the reasons described above.

'A' is wrong: -161" to +12.5", per T-101 (RPV Control). Plausible if the examinee correctly determines that no ATWS conditions exist based on APRM Downscale lights, however, the examinee may determine that due to the RPV level below +12.5", step RC/L-5 applies, which states that if RPV level cannot be restored and maintained above +12.5" then maintain RPV level above -161". Based on that the maximum level band changes to +12.5" to -161".

'B' is wrong: -186" to +10", per T-117 (Level/Power Control). Plausible if the examinee determines that ATWS conditions exist based on indications of 2 rods remaining full out, and reactor power unknown. Based on that, the examinee also determines that RPV level needs to be deliberately lowered to -50" and as a result step LQ-15 states that if RPV level was deliberately lowered then RPV level needs to be maintained between -186" to level to which it was lowered from (+10").

'C' is wrong: +12.5" to +54", per T-101 (RPV Control). Plausible based on RPV level below +12.5", T-101, RPV Control is entered, and ATWS condition exists but reactor power is less than 4% power due to indications that APRM downscale lights are lit. T-101, Step RC/L-2 ATWS bases states that if operators have positive confirmation that the reactor is, and will remain, shutdown under all conditions without boron, an ATWS is NOT in progress. This determination is best obtained by determining that no control rod is withdrawn beyond the maximum subcritical banked withdrawal position (MSBWP, position 02). However, other criteria can also be used to demonstrate that the reactor will remain shutdown under all conditions, without boron. Also, Caution for RC/Q states that APRM downscale may be used to ensure reactor power is less than 4%. Therefore, RC/L-4 directs operators to restore and maintain RPV level between +12.5" and +54" when there is an ATWS condition with reactor power less than 4%.

Question 87 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	2.50
System ID:	1104205
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LLOT1560
Topic:	(SRO) Scram/ATWS - Determine Applicable RPV Water Level Band
Num Field 1:	4.2
Num Field 2:	4.2
Text Field:	295006 AA2.03
Comments:	Level: SRO Tier: 1 Group: 1 KA # and Rating: 295006 AA2.03 (4.0/4.2) KA Statement: 295006 SCRAM Ability to determine and/or interpret the following as they apply to SCRAM: AA2.03 Reactor water level References: T-101, Rev.21 T-117, Rev.17 Examinee References: None Learning Objective: LLOT1560 Question source: LGS 2012 ILT NRC Exam, Question #77 Question history: LGS 2012 ILT NRC Exam, Question #77 Cognitive level: Higher 10 CFR 55: 43.5 Comments: None

88 **ID: 1104206** **Points: 1.00**

SRO

Plant conditions:

- Reactor level is 35", steady
- Reactor pressure is 500 psig
- Drywell pressure is 12 psig
- Drywell temperature is 275°F
- Suppression Pool pressure is 10 psig
- Suppression Pool temperature is 115°F
- Suppression Pool level is 23.5 feet

WHICH ONE of the following actions should be given the **HIGHEST** priority?

(Assume that none of the suggested actions has yet been performed.)

A. Spray the Drywell per T-225

B. Lower Suppression Pool level per T-102

C. Spray the Suppression Pool per T-225

D. Perform Emergency Blowdown per T-112

Answer: C

Answer Explanation

The SRO (ILT) examinee is expected to answer this question from memory by recognizing/recalling/interpreting certain parameter values, such as: 1) the combination of a 23.5 ft supp pool level with a 115°F supp pool temperature is ALWAYS acceptable when evaluated against any of the T-102 Curves; 2) a supp pool level of 23.5', although higher than normal, does not yet require level to be lowered; 3) a 10 psig supp pool pressure is no where near being a threat to PCPL; 4) a 12 psig drywell pressure with a 275°F drywell temperature is well on the SAFE side of the DWSIL curve; 5) none of the given parameter values, in and of themselves, are near an Emergency Blowdown limit. The examinee is also expected to recall that although the supp pool should have been sprayed before reaching 7.5 psig (see T-102, PC/P leg), that action is, nonetheless, still to be performed BEFORE proceeding thru the STOP sign of step PC/P-7 (i.e., BEFORE spraying the DW). Therefore, the required action with the HIGHEST priority (given the answer choices, none of which has yet been performed) must be to spray the supp pool (as directed by T-102, step PC/P-5, and performed using T-225).

'C' is correct: Spray the Suppression Pool per T-225. Correct for the reasons described above.

'A' is wrong: Spray the Drywell per T-225. Plausible for the reasons described above. This examinee forgets that the supp pool is to be sprayed BEFORE the drywell.

'B' is wrong: Lower Suppression Pool level per T-102. Plausible for the reasons described above. This examinee is distracted by the much-higher-than-normal supp pool level.

'D' is wrong: Perform Emergency Blowdown per T-112. Plausible to the examinee who cannot effectively interpret any of the critical stem conditions information and, so, defaults to believing that an Emergency Blowdown is a priority.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 88 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	4	
Difficulty:	2.50	
System ID:	1104206	
User-Defined ID:	REV 01, 12/15/14	
Cross Reference Number:	LLOT1560.6	
Topic:	(SRO) T-102 - Determine Required Action Given PC Parameters	
Num Field 1:	3.9	
Num Field 2:	4.2	
Text Field:	295010 2.1.25	
Comments:	Level	SRO
	Tier	1
	Group	2
	KA # and Rating	295010 2.1.25* (3.9/4.2)
	KA Statement	295010 High Drywell Pressure 2.1.25 Ability to interpret reference materials, such as graphs, curves, tables,etc.
	References	T-102, Rev.24 T-225 U/1, Rev.22
	Examinee References	None
	Learning Objective	LLOT1560.6
	Question source	Bank 554360
	Question history	None
	Cognitive level	Higher
	10 CFR 55	43.5

	Comments	*Justification for KA match: Although the correct involves "spraying the supp pool" (rather than the drywell), the T-102 action to spray the pool is a part of the same mitigation strategy as is spraying the drywell; in fact, both are part of executing the PC/P (Primary Containment Pressure) leg of T-102. ALSO - While no "Examinee References" are provided for this question, it nonetheless involves the requisite "ability to interpret...graphs, curves...etc.", based on the Examinee having to recall the essential characteristics of the HCTL, PCPL, and DWSIL curves of T-102. The LGS Candidates have been trained/tested in this fashion throughout their ILT training program. --- Editorial (added to the "Justification for KA match") to address NRC Rev.00 comments.

89 **ID: 1104267** **Points: 1.00**

SRO

Unit 1 is in a SITE AREA EMERGENCY, with the following:

- Complete loss of all offsite power and LOCA
- The only Unit 1 4KV Safeguard Buses being powered are D11 and D12
- RPV water level is -165", steady
- Drywell radiation monitor is reading 30 R/hr
- Actual offsite dose is 200 mRem TEDE and 300 mRem CDE Thyroid

WHICH ONE of the following will, **by itself**, escalate Unit 1 to a GENERAL EMERGENCY?

A. Loss of D11 DG or D12 DG

B. RPV water drops to -182", steady

C. Drywell radiation monitor reading 95 R/hr

D. Actual offsite dose > 1000 mRem TEDE

Answer: D

Answer Explanation

Refer to EP-AA-1008, LGS EAL "Hot Matrix".

'D' is correct: Actual offsite dose > 1000 mRem TEDE. Per EAL RG1, Threshold Criteria #2, an actual offsite dose of > 1000 TEDE warrants a GENERAL EMERGENCY.

'A' is wrong: Loss of D11 DG or D12 DG. Given the stem conditions, with respect to "Loss of AC Power", Unit 1 is only at the UNUSUAL EVENT level (per EAL MU1). Were either the D11 DG or D12 DG to be lost, the resulting "Loss of AC Power" emergency classification would be an ALERT (per MA1). Plausible to the inexperienced SRO Candidate who recalls only that a single remaining 4KV bus (which would be the result of losing either of the two DGs) places the Unit one power source away from a "Station Blackout" situation; often, such Candidates wrongly conclude that this alone is reason to declare a G.E.

'B' is wrong: RPV water drops to -182", steady. -182" (steady) is still above the -186" threshold value of the "loss of Fuel Clad" portion of the FP Barrier Matrix. Plausible, again, to most inexperienced SRO Candidates.

'C' is wrong: Drywell radiation monotor reading 95 R/hr. A 95 R/hr drywell radiation level is still well below the 190 R/hr threshold for a "loss of the Fuel Clad" (per the "Fuel Clad" portion of the Fission Product Barrier Matrix; see EAL FG1). Plausible to the inexperienced SRO Candidate who incorrectly recalls the ">190 R/hr" threshold value, associated with the "loss of Fuel Clad", as >90 R/hr, instead.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 89 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	5	
Difficulty:	3.00	
System ID:	1104267	
User-Defined ID:	REV00, 11/17/14	
Cross Reference Number:		
Topic:	(SRO) Recognize degrading plant condition warranting declaration of a General Emergency	
Num Field 1:	2.9	
Num Field 2:	4.2	
Text Field:	295017 AA2.01	
Comments:	Level	SRO
	Tier	1
	Group	2
	KA # and Rating	295017 AA2.01 (2.9/4.2)
	KA Statement	295017 High Offsite Release Rate** Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: AA2.01 †Off-site release rate
	References	EP-AA-1008 Hot Matrix, Rev.26
	Examinee References	None
	Learning Objective	No specified objective
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	43.5
	Comments	**Justification for KA match: The description of the RG1 classification clearly associates a total dose (in this case, >1000 mR TEDE or 5000 mR Thyroid CEDE) projected to be received for the duration of a release. In other words, it relates a dose to a time factor. As such, although the RG1 units of measure are dose rather than dose rate, the intent of the 295017 KA is satisfied by the an application to this EAL.

90 **ID: 1104304** **Points: 1.00**

SRO

Unit 1 is operating at 100% power.

Operators determine the following:

- AGAFs are reading 1.03 on APRMs 1 and 3
- The AGAFs will not respond to adjustment

WHICH ONE of the following describes the Tech Spec required action?

A. Be in STARTUP within 18 hours

B. Be in STARTUP within 12 hours

C. Insert a half-scram on either RPS 'A' or 'B' within 6 hours

D. Place APRM channels 1 and 3 in a tripped condition within 12 hours

Answer: A

Answer Explanation

Refer to U/1 Tech Spec 3.3.1 (RPS), Table 4.3.3.1-1, Functional Units 2.b and 2.c, as well as Note (d). Also refer to Table 3.3.1-1, Functional Units 2.b and 2.c, as well as Note (m). A review of these Tables and Notes reveals the following:

- All 4 APRM channels (1, 2, 3, 4) input to both RPS Trip Systems (A and B)
- A "Mimimum" of 3 of the channels must be OPERABLE for each of the two Trip Systems
- The "bad" AGAF (see Note (d)) affects Functional Units 2.b and 2.c

Given the stem conditions, we conclude the following:

- Only two APRM channels (2 and 4) remain OPERABLE, for each Trip System, for each of the two Functional Units (2.b, 2.c)
- Thus, each Trip System has one OPERABLE channel less than the three required

A review of ACTION 'a' and the (*) Note at the bottom of the page reveals the following required action:

- Within 1 hour, verify there is at least two OPERABLE channels in each Trip System
- Because we have the two OPERABLE channels, nothing further is required by ACTION 'a'

However, ACTION 'b', and the (**) Note at the bottom of the page, requires the following:

- Within 12 hours, APRMs 1 and 3 must be placed in a tripped condition
- If we were to do this, a full scram would occur

Fortunately, ACTION 'd' provides for an option in lieu of intentionally scramming the plant:

- Within 12 hours (action time associated with ACTION 'b'), initiate ACTION "4" (as noted for Functional Units 2.b/2.c on Table 3.3.1-1)

Regarding ACTION 4...

- By itself, Action 4 would require the Mode Switch to be in STARTUP within 6 hours
- However, as already discussed above, we do not have to initiate that ACTION until the expiration of the 12 hour action time permitted by Action 'b'

From all of this, we find that the "Tech Spec required action" for this event is as follows:

- Be in STARTUP within 18 hours (i.e., the 12 hours of Action 'b' + the 6 hours of Action 4)

'A' is correct: Be in STARTUP within 18 hours. Correct for the reasons described above.

'B' is wrong: Be in STARTUP within 12 hours. Plausible to the examinee who does know how to analyze the two inoperable APRMS (as we've done above), but who fails to recognize that ACTION 'd' extends the Action '4' time to 18 hours, or who simply fails to recognize the "initiate" term used in the wording of ACTION 'd' (i.e., he/she wrongly substitutes the 12 hour time of ACTION 'b' for the 6 hours of ACTION 4).

'C' is wrong: Insert a half-scram on either RPS 'A' or 'B' within 6 hours. This choice suggests ACTION 'c'. It's plausible to the examinee who fails to recognize that the (**) Note excludes Functional Units 2.b and 2.c from ACTION 'c'.

'D' is wrong: Place APRM channels 1 and 3 in a tripped condition within 12 hours. This choice suggests that we can in fact perform ACTION 'b' without causing a full scram. This is very plausible to the examinee who does not fully understand how the "Two-out-of-Four" VOTER arrangement works for the Power Range Neutron Monitoring System. As such, that examinee believes that Channels 1 and 3 follow the "usual" convention of being associated only with RPS Trip System 'A' (where, for example, Channels 2 and 4 would similarly be associated with RPS Trip System 'B'). If that were true, then simply inserting a half-scram (by tripping the APRM channels) would be a logical action to take.

Question 90 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	6	
Difficulty:	3.00	
System ID:	1104304	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	LLOT0071.12	
Topic:	(SRO) Determine Tech Spec Action for two inoperable APRMs	
Num Field 1:	4.0	
Num Field 2:	4.7	
Text Field:	212000 2.2.22	
Comments:	Level	SRO
	Tier	2
	Group	1
	KA # and Rating	212000 2.2.22 (4.0/4.7)
	KA Statement	212000 RPS 2.2.22 Knowledge of limiting conditions for operations and safety limits.
	References	U/1 Tech Spec 3.3.1

	Examinee References Learning Objective Question source Question history Cognitive level 10 CFR 55 Comments	U/1 Tech Spec 3.3.1 (***EXCLUDING the following:*** All of Table 3.3.1-2; and the second page (that includes the TSV/TCV logic) of Table 3.3.1-1) LLOT0071.12 New None Higher 43.2 None

91 **ID: 1104380** **Points: 1.00**

SRO

Unit 1 is operating at 100% power when the following occurs:

- 'A' and 'C' Narrow Range Level Indicators (at 10C603) simultaneously fail hard downscale
- The failures are due to the associated level transmitter outputs failing to 0 mA

WHICH ONE of the following describes:

(1) the response of reactor power, and
(2) the Tech Spec implication?

A. (1) Lowers to about 65%
(2) Must restore either the 'A' or 'C' level instrument to OPERABLE within 72 hours

B. (1) Remains unchanged
(2) Must restore either the 'A' or 'C' level instrument to OPERABLE within 72 hours

C. (1) Lowers to about 65%
(2) Because the instruments have failed downscale, the required Tech Spec action is already satisfied

D. (1) Remains unchanged
(2) Because the instruments have failed downscale, the required Tech Spec action is already satisfied

Answer: A

Answer Explanation

Refer to S06.1.H U/1, Responding to Alarms at the FWLCS Operator Station, Attachment 1 ALARM LIST, page 8 of 27, "Signal Identity" 1XX-FW300.ILE. A review of the "Automatic Actions" for this item shows that the simultaneous failure of two RPV level transmitters (such as the two identified in the the question stem) results in an automatic Recirc Pump runback to 28% speed due to a "sensed" RPV level of < 12.5". SRO examinee is expected to recognize that reactor power will lower and stabilize at about 65% as a consequence of the runback.

The examinee is expected to recognize that these two failed RPV level indicators (LI-42-1R606A and C) input only to FWLCS; they are not used by RPS. As such, the "RPS Instrumentation" Tech Spec (3.3.1) does not apply, but the "Feedwater/Main Turbine Trip Actuation Instrumentation" Tech Spec (3.3.9) does apply. Note - the two failed level instruments are among the four (total), including the 'B' and 'D' instruments, that make up the main turbine and reactor feed pump turbine trip logic when RPV level rises to +54". Per Tech Spec Table 3.3.9-1, all four of the instruments are required to be OPERABLE. The two failed (inoperable) instruments make ACTION 'c', therefore, the required action. That ACTION translates to the following: "Restore either the 'A' or 'C' level instrument to OPERABLE within 72 hours.

Suppose these particular two instruments ('A' and 'C') were, instead, among the four (total) instruments used by RPS (for the RPV Low Level +12.5" scram function). The RPS Trip System association with its Narrow Range level instruments is as follows: (A or C) AND (B or D). In that case, only RPS Trip System 'A' would actuate in response to the "sensed" RPV low level; i.e., an RPS 'A' half-scram would result. Refer to RPS Tech Spec 3.3.1, where Table 3.3.1-1, Functional Unit 4 shows that both level instruments are required to be OPERABLE for each Trip System. In this case, ACTION 'b' would apply; however, with RPS 'A' already "tripped" (i.e., the half-scram), that ACTION is already completed and no additional ACTION is required by the LCO.

'A' is correct: (1) Lowers to about 65%; (2) Restore either the 'A' or 'C' level instrument to OPERABLE within 72 hours. Correct for the reasons described above.

'B' is wrong: (1) Remains unchanged; (2) Restore either the 'A' or 'C' level instrument to OPERABLE within 72 hours. Part (1) is very plausible because the examinee is likely not to recognize/recall the association between the level insturment downsclae failure and the resulting Recirc Pump 28% Runback.

'C' is wrong: (1) Lowers to about 65%; (2) Because the instruments have failed downscale, the required Tech Spec action is already satisfied. Part (2) is plausible to the examinee who believes that these two level instruments are used by RPS (as discussed above).

'D' is wrong: (1) Remains unchanged; (2) Because the instruments have failed downscale, the required Tech Spec action is already satisfied. Plausible for the same reasons as that for choices 'A' and 'C'.

Question 91 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	3.00
System ID:	1104380
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LLOT0540,16
Topic:	(SRO) FWLCS LT failures - Predict Plant Response and Tech Spec Implication
Num Field 1:	3.6
Num Field 2:	3.7
Text Field:	259002 A2.03
Comments:	Level SRO Tier 2 Group 1 KA # and Rating 259002 A2.03 (3.6/3.7)

	KA Statement	259002 Reactor Water Level Control Ability to (a) predict the impacts of the following on the REACTOR WATER LEVEL CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.03 Loss of reactor water level input
	References	S06.1.H U/1, Rev.14 U/1 Tech Spec 3.3.1 (latest) U/1 Tech Spec 3.3.9 (latest)
	Examinee References	None
	Learning Objective	LLOT0540, 16
	Question source	New
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.5, 43.2
	Comments	None

92 **ID: 1118691** **Points: 1.00**

SRO

Unit 2 is in OPCON 3.

WHICH ONE of the following describes the **EARLIEST** point at which an Emergency Action Level declaration is required for "Loss of Vital DC Power"?

A. Less than 115 VDC on 125 VDC buses 2FA and B for greater than or equal to 15 minutes

B. Less than 105 VDC on 125 VDC buses 2FA and B for greater than or equal to 15 minutes

C. Less than 115 VDC on 125 VDC buses 2FA, B, C, and D for greater than or equal to 15 minutes

D. Less than 105 VDC on 125 VDC buses 2FA, B, C, and D for greater than or equal to 15 minutes

Answer: D

Answer Explanation

Refer to Hot Matrix for "DC Power". There we find that only a Site Area Emergency (SAE); i.e., MS3, Threshold exists, and that threshold requires the loss of ALL FOUR vital 125 VDC buses (FA, B, C, and D; i.e., Div 1, 2, 3, and 4); the "loss" is defined as <105 VDC for >/=15 minutes.

'D' is correct: Less than 105 VDC on 125 VDC buses 2FA, B, C, and D for greater than or equal to 15 minutes. Correct for the reasons described above.

'A' is wrong: Less than 115 VDC on 125 VDC buses 2FA and B for greater than or equal to 15 minutes. This choices suggests that anything less than 115 VDC qualifies as a lost bus. It also suggests that only the Div 1 and Div 2 buses are of a concern with repsect to the EAL. In a closed-reference setting, this choices is very plausible to the Examinee who associates the loss of Div 1 and Div 2, alone, as being sufficient to disable the two important RHR subsystems ('A' and 'B') that are necessary to mitigate Primary Containment high temperature/high pressure emergency conditions; i.e....the need to cool the suppression pool and/or spray the pool/drywell.

'B' is wrong: Less than 105 VDC on 125 VDC buses 2FA and B for greater than or equal to 15 minutes. Although this choice does identify the correct voltage threshold (<105 VDC), it too suggest that only the Div 1 and Div 2 buses are of concern, Plausible for reasons similar to that for choice 'A'.

'C' is wrong: Less than 115 VDC on 125 VDC buses 2FA, B, C, and D for greater than or equal to 15 minutes. Plausible to the Examinee who cannot recall the voltage threshold that defines a "loss" of the DC bus.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 92 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.50	
System ID:	1118691	
User-Defined ID:	REV.01, 12/15/14	
Cross Reference Number:	NONE	
Topic:	(SRO) Recall EAL for Loss of Vital DC Power	
Num Field 1:	2.9	
Num Field 2:	4.6	
Text Field:	263000 2.4.41	
Comments:	Level	SRO
	Tier	2
	Group	1
	KA # and Rating	263000 2.4.41 (2.9/4.6)**
	KA Statement	263000 DC Electrical Distribution 2.4.41 Knowledge of the emergency action level thresholds and classifications.
	References	EA-AA-1008, Rev.26 - Hot Matrix
	Examinee References	None
	Learning Objective	No specified objective
	Question source	New**
	Question history	None
	Cognitive level	Lower
	10 CFR 55	43.5
	Comments	**This is a "New" question to replace "old"(Vision ID: 1104381) to address NRC Rev.00 comments. KA swap was necessary to support NRC concerns for "excessive number of questions related to Tech Specs."

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

93 **ID: 1104389** **Points: 1.00**

SRO

Unit 1 is operating at 100% power, with the following:

- Drywell cooling is maximized
- MCC D114-R-G is tagged out for repairs
- Drywell air temperature is 125°F

A loss of MCC D134-R-E occurs; as a result, drywell air temperature begins to rise.

fre 3/10/15 Delete 893

Assume drywell air temperature rises at a constant rate of 5°F/hr for each drywell fan that is no longer operating.

WHICH ONE of the following identifies:

(1) the approximate time it will take for drywell air temperature to exceed its Tech Spec limit, and
(2) the Tech Spec time allowed for restoring drywell air temperature to within its Tech Spec limit?

A. (1) 1 hour
(2) 8 hours

B. (1) 2 hours
(2) 8 hours

C. (1) 1 hour
(2) 24 hours

D. (1) 2 hours
(2) 24 hours

Answer: B

Answer Explanation

All operators and ILT Candidates understand that the phrase "drywell cooling is maximized" means only the following: All 8 Unit Coolers are in service, with 1 fan operating in each of the Unit Coolers, with 1 Drywell Chilled Water (DWCW) Chiller operating and 2 DWCW Circ Pumps operating.

The following identifies the power supplies for the drywell unit cooler fans:

- MCC D114-R-G feeds 1A1V212, 1C1V212, 1E1V212, 1G1V212 (a total of 4 fans)
- MCC D124-R-G feeds 1B1V212, 1D1V212, 1F1V212, 1H1V212 (a total of 4 fans)
- MCC D134-R-E feeds 1A2V212, 1E2V212 (a total of 2 fans)
- MCC D134-R-H feeds 1C2V212, 1G2V212 (a total of 2 fans)
- MCC D144-R-H feeds 1D2V212, 1F2V212 (a total of 2 fans)
- MCC D144-R-E feeds 1B2V212, 1H2V212 (a total of 2 fans)

NOTE - all of these power supplies can be validated by reviewing 1S77.1.A (COL), pages 1 - 6.

Stem conditions indicate that MCC D114-R-G is out of service, yet "drywell cooling is maximized." this necessarily means that the fans fed from MCC's D134-R-E (2 fans) and D134-R-H (2 fans) are operating in place of the 4 fans that are not operating because of the MCC outage.

Rev 3/10/15
Delete Q93

When MCC D134-R-E is lost, two fans trip (they are 1A2V212 and 1E2V212). The stated "assumption" that drywell air temperature rises at a constant rate of 5°F/hr for each fan that has been lost (2 fans) means that the air temperature is rising at a rate of 10°F/hr.

The Tech Spec 3.6.1.7 Drywell Average Air Temperature limit is 145°F. Therefore, starting at a temperature of 125°F, it will take 2 hours to reach the Tech Spec limit.

Per the TS 3.6.1.7 ACTION, operators have 8 hours to restore the temperature to within the limit.

'B' is correct: (1) 2 hours; (2) 8 hours. Correct for the reasons descrubed above.

'A' is wrong: (1) 1 hour; (2) 8 hours. Part (1) assumes a total of 4 fans are lost when MCC D134-R-E is lost. This is plausible to the examinee who fails to recall that the Division 3 and Division 4 associated MCCs (i.e., D134 and D144, respectively) each power only two fans, as opposed to the Div 1 and Div 2 MCCs (i.e., D114 and D124, respectively) that each power 4 fans.

'C' is wrong: (1) 1 hour; (2) 24 hours. Part (1) is plausible for the same reasons as that for choice 'A'. Part (2) suggests the restoration time allowed by Tech Spec 3.6.2.1, ACTION 'b' when the 95°F suppression pool temperature limit is exceeded; plausible for that reason.

'D' is wrong: (1) 2 hours; (2) 24 hours. Part (2) is plausible for the same reason as that for choice 'C'.

Question 93 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	1104389
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	NONE
Topic:	(SRO) Predict DW air temperature rise time and recall Tech Spec AOT to restore to within limits
Num Field 1:	3.6
Num Field 2:	3.8
Text Field:	223001 A2.10
Comments:	Level SRO Tier 2 Group 2 KA # and Rating 223001 A2.10 (3.6/3.8)

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

	KA Statement	223001 Primary CTMT and Auxiliaries Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.10 High drywell temperature
	References	1S77.1.A (COL), Rev.3 U/1 Tech Spec 3.6.1.7 (latest) U/1 Tech Spec 3.6.2.1 (latest)
	Examinee References	None
	Learning Objective	No specified objective
	Question source	New
	Question history	None
	Cognitive level	Higher
	10 CFR 55	41.5, 43.2
	Comments	None

Pro 3/10/15
Delete Q93

94 **ID: 1104465** **Points: 1.00**

SRO

Unit 2 is operating at 100% power.

WHICH ONE of the following Control Rod problems would require a plant shutdown within 12 hours upon entry into the associated Tech Spec?

A. With 7 rods already inoperable, and 8th rod is declared inoperable

B. The average scram time for the 3 fastest rods in a 2-by-2 array is twice the allowable limit at all positions of insertion

C. The average scram insertion time for all OPERABLE control rods exceeds the limit only for insertion from position 48 to position 39

D. Two accumulators (for withdrawn rods) are declared inoperable due to internal leakage

Answer: C

Answer Explanation

'C' is correct: The average scram insertion time for all OPERABLE control rods exceeds the limit only for insertion from position 48 to position 39. Per Tech Spec 3.1.3.3, exceeding the limit the scram time limit at any point of insertion (39, 25, or 05) requires the ACTION to "be in HOT SHUTDOWN within 12 hours."

'A' is wrong: With 7 rods already inoperable, an 8th rod is declared inoperable. Per Tech Spec 3.1.3.1, ACTION 'c', the shutdown isn't required until a 9th rod is declared inoperable. Plausible to the examinee who does recall something about "8" rods being inoperable, but can't recall whether the limit is 8 or if the ACTION is required when the 8th becomes inoperable.

'B' is wrong: The average scram time for the 3 fastest rods in a 2-by-2 array is twice the allowable limit at all positions of insertion. Per Tech Spec 3.1.3.2, exceeding the scram time limit at any point of insertion (regardless of the degree to which it is exceeded) requires the associated ACTION. However, per that ACTION, continued plant operation is permitted for at least 60 days. Plausible to the examinee who is distracted by what appears to be an unacceptably long scram time ("twice the limit").

'D' is wrong: Two accumulators (for withdrawn rods) are declared inoperable due to internal leakage. Per Tech Spec 3.1.3.5, ACTION a.2 applies. Because the accumulators are inoperable due to internal leakage problems, and not because of a loss of CRD charging water header pressure, the only action is declare the two associated rods inoperable. Plausible to the examinee who distracted by the fact that the accumulator are for withdrawn rods.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 94 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	2.50
System ID:	1104465
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LLOT0060.12
Topic:	(SRO) Recognize control rod problem requiring plant shutdown within 12 hours upon Tech Spec entry
Num Field 1:	4.0
Num Field 2:	4.7
Text Field:	201003 2.2.22
Comments:	Level SRO Tier 2 Group 2 KA # and Rating 201003 2.2.40 (3.4/4.7) KA Statement 201003 Control Rod Drive Mechanisms 2.2.22 Knowledge of limiting conditions for operations and safety limits. References U/2 Tech Spec 3.1.3.1 (latest) U/2 Tech Spec 3.1.3.2 (latest) U/2 Tech Spec 3.1.3.3 (latest) U/2 Tech Spec 3.1.3.4 (latest) Examinee References None Learning Objective LLOT0060.12 Question source New Question history None Cognitive level Lower 10 CFR 55 43.2 Comments None

95 **ID: 1118715** **Points: 1.00**

SRO

Unit 1 LOCA is in progress, with the following:

- Drywell H_2 is 7%
- Drywell O_2 is unknown
- Suppression Pool H_2 is 5%
- Suppression Pool O_2 is 4%
- Drywell Fans are NOT running
- Post-LOCA Recombiners are NOT running
- CY-LG-130-009 (Determination of Gaseous Effluent Rad Monitor Setpoints) has NOT been completed

WHICH ONE of the following describes the required action per T-102?

A. Inert/Purge Suppression Pool using maximum flowrate per T-228, exceeding offsite release rate limits if necessary

B. Inert/Purge Drywell using maximum flowrate per T-228, exceeding offsite release rate limits if necessary

C. Inert Suppression Pool using N_2 (low flow mode) per T-228

D. Inert Drywell using N_2 (low flow mode) per T-228

Answer: B

Answer Explanation

Refer to T-102, Sheet 2 of 2, specifically Tables PC/G-1 and 2. Stem conditions indicate that DW hydrogen is 7% and DW oxygen in unknown. Per Table PC/G-1, this combination requires operators to execute the DW/G-3 leg. Step DW/G-3.3 requires that the DW fans be secured (they are NOT currently running) and the Post-LOCA Recombiners be secured (they are NOT currently running) before proceeding to Step DW/G-3.4 which directs operators to then Inert/Purge the DW using the maximimum flowrate per T-228 exceeding offsite release rate limits if necessary.

NOTE - Although this question appears, at first, to be far beyond the capability of an SRO Candidate to answer from-memory (in terms of the complexity of the T-102 Tables PC/G-1 and 2), it really only relies on the examinee recognizing the significance of the >6% H2 and UNKNOWN O2 combination in the drywell...identifying the threshold at which the mitigation strategy must be as aggressive as is possible..i.e., inerting or purging the DW at the maximum available flowrate while exceeding the offsite release rate limits if necessary. In other words, the SRO Candidate need only translate the stem conditions to the sole action (of the 4 actions suggested by the answer choices) that will most aggressively protect the structural integrity of the Primary Containment (being threatened by the extremely high gas concentrations).

'B' is correct: Inert/Purge Drywell using maximum flowrate per T-228, exceeding offsite release rate limits if necessary. Correct for the reasons described above.

'A' is wrong: Inert/Purge Suppression Pool using maximum flowrate per T-228, exceeding offsite release rate limits if necessary. This choice suggests the action of Step SP/G-3.5; however, with the given supp pool 5% hydrogen and 4% oxygen, Table PC/G-2 direct operators to execute the SP/G-1 leg, not the SP/G-3 leg. With respect to plausibility, please see the **NOTE** above.

'C' is wrong: Inert Suppression Pool using N2 (low flow mode) per T-228. This choice suggests the action of Step SP/G-1.9; however, as already discussed for choice 'A', operators will not be executing the SP/G-1 leg. What's more, with the CY-LG-130-009 procedure NOT yet completed (as indicated in the stem conditions), operators cannot proceed past the WHEN-THEN condition of Step SP/G-1.7. With respect to plausibility, please see the **NOTE** above.

'D' is wrong: Inert Drywell using N2 (low flow mode) per T-228. This choice suggests the action of Step DW/G-1/9; however, as already discussed, operators will not be executing the DW/G-1 leg. What's more, with the CY-LG-130-009 procedure NOT yet completed (as indicated in the stem conditions), operators cannot proceed past the WHEN-THEN condition of Step DW/G-1.7. With respect to plausibility, please see the **NOTE** above.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 95 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	2.50
System ID:	1118715
User-Defined ID:	REV 02, 01/06/15
Cross Reference Number:	NONE
Topic:	(SRO) Determine NEXT T-102 Action based on DW hydrogen concentration
Num Field 1:	3.7
Num Field 2:	4.7
Text Field:	500000 2.4.6
Comments:	Level: SRO Tier: 1 Group: 2 KA # and Rating: 500000 2.4.6 (3.7/4.7)** KA Statement: 500000 High CTMT Hydrogen Concentration 2.4.6 Knowledge of EOP mitigation strategies. References: T-102, Rev.24 Examinee References: None Learning Objective: No specified objective Question source: New** Question history: None Cognitive level: Higher 10 CFR 55: 43.5 Comments: **Replacement "new" question to address NRC Rev.00 comments. Comment resolution required a KA swap as well.

96 **ID: 1104467** **Points: 1.00**

SRO

LGS has declared a GENERAL EMERGENCY.

Command & Control has been transferred to the TSC.

Consider the following ERO personnel:

1. Shift Manager / Shift Emergency Director
2. Radiation Protection Manager
3. Station Emergency Director

WHICH ONE of the following identifies (from the above list) **ALL** of those who can **APPROVE** an Emergency Exposure personnel dose?

A. 1, 3

B. 2, 3

C. 2

D. 3

Answer: D

Answer Explanation

Refer to EP-AA-113-F-02 (Authorization for Emergency Exposure). The Shift Manager / Shift Emergency Director may approve prior to transferring Command and Control to the Station Emergency Director. After the transfer of command and control, only the Station Emergency Director can approve the emergency exposure.

'D' is correct: 3. Correct for the reasons described above.

'A' is wrong: 1, 3. Plausible to the examinee who recognizes that the Shift Manager is always the initial "Emergency Director" and so believes that he/she can still approve the exposure even after transferring command and control.

'B' is wrong: 2, 3. Plausible to the examinee who recalls that the Radiation Protection Manager (RPM) is involved in the process (as shown on EP-AA-113-F-02) but who forgets that the RPM only "reviews" the exposure request; he/she cannot approve the request.

'C' is wrong: 2. Plausible to the examinee who recognizes that the Station Emergency Director has overall command and control of the ERO, but who believes that the expertise that only the RPM possesses is necessary to ultimately approve the exposure request.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 96 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	1104467
User-Defined ID:	REV 01, 12/15/14
Cross Reference Number:	NONE
Topic:	(SRO) Recall who can Approve an Emergency Exposure request
Num Field 1:	3.2
Num Field 2:	3.7
Text Field:	2.3.4
Comments:	Level: SRO Tier: 3 Group: N/A KA # and Rating: 2.3.4 (3.2/3.7) KA Statement: 2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions. References: EP-AA-113-F-02, Rev.B Examinee References: None Learning Objective: No specified objective Question source: New Question history: None Cognitive level: Lower 10 CFR 55: 43.4 Comments: Editorial changes to answer choices to address NRC Rev.00 comments.

97 **ID: 1104468** **Points: 1.00**

SRO

LGS has declared an ALERT and has completed all required notifications.

At 1300 hours:

- SITE AREA EMERGENCY (SAE) is declared

At 1313 hours:

- GENERAL EMERGENCY (GE) is declared
- SAE notifications have NOT yet been transmitted

WHICH ONE of the following describes the State/Local notification requirements for the situation above?

A. Do not make the SAE notification; instead, complete the GE notification no later than 1328 hours.

B. Do not make the SAE notification, but do notify State/Locals that the event has been escalated. Then complete the GE notification no later than 1328 hours.

C. If possible, complete the GE notifcation (in lieu of the SAE) no later than 1315 hours. Otherwise, complete the SAE notifcation no later than 1315 hours and the GE notification no later than 1328 hours.

D. Complete the SAE notification no later than 1328 hours, then complete the GE notification no later than 1343 hours.

Answer: C

Answer Explanation

Refer to EP-AA-111, section 4.1, 2nd NOTE, which reads: "If a higher classification is made prior to transmitting an event notification, then notification for the higher classification can supercede the previous event notification, provided that it can be performed within the 15-minute timeframe of the previous event"..."IF the notification of a higher classification cannot be performed within the 15-minute timeframe of the previous event, then the previous event notification is required within its 15-minute timeframe, and the subsequent event notification is required within its 15-minute timeframe.

'C' is correct: If possible, complete the GE notifcation (in lieu of the SAE) no later than 1315 hours. Otherwise, complete the SAE notifcation no later than 1315 hours and the GE notification no later than 1328 hours. Correct for the reasons described above.

'A' is wrong: Do not make the SAE notification; instead, complete the GE notification no later than 1328 hours. Plausible to the inexperienced SRO Candidate who believes that each EAL notification should be completed before regarding the escalated EAL.

'B' is wrong: Do not make the SAE notification, but do notify State/Locals that the event has been escalated. Then complete the GE notification no later than 1328 hours. Plausible to the inexperienced SRO Candidate who believes that although the actual SAE notification is not required (because it has already escalated to a GE), nonetheless, offsite authorities should be made made aware in advance so as to provide them an opportunity to plan for the GE.

'D' is wrong: Complete the SAE notification no later than 1328 hours, then complete the GE notification no later than 1343 hours. Plausible to the inexperienced SRO Candidate who recalls only the 15-minute notification requirements and believe those notifications should be made in the order of the declared EALs.

Question 97 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.50	
System ID:	1104468	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	NONE	
Topic:	(SRO) Recall Time Limits for EAL Notification	
Num Field 1:	2.4	
Num Field 2:	4.4	
Text Field:	2.4.38	
Comments:	Level	SRO
	Tier	3
	Group	N/A
	KA # and Rating	2.4.38 (2.4/4.4)
	KA Statement	2.4.38 Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required.
	References	EP-AA-111, Rev.18
	Examinee References	None
	Learning Objective	No specified objective
	Question source	Bank 985666
	Question history	None
	Cognitive level	Lower
	10 CFR 55	43.5
	Comments	None

98 **ID: 1104484** **Points: 1.00**

SRO

Unit 1 is operating at 100% power.

WHICH ONE of the following identifes how often the Unit 1 CRS is required to perform a **solo** walkdown of MCR panels, per OP-AA-103-102 (Watchstanding Practices)?

A. Every 2 hours

B. Every 4 hours

C. Every 6 hours

D. Once per shift

Answer: B

Answer Explanation

Per OP-AA-103-102, step 4.4.2.1: every 4 hours (solo). Per step 4.4.2.2, a paired (with the RO) walkdown is performed once per shift.

'B' is correct: Every 4 hours. Correct for the reasons described above.

'A' is wrong: Every 2 hours. Plausible to the examinee who does not recall the OP-AA-103-102 requirement.

'C' is wrong: Every 6 hours. This suggests a "twice per shift" requirements. Plausible to the examinee who does not recall the OP-AA-103-102 requirement.

'D' is wrong: Once per shift. Plausible to the examinee who believes that the "Command and Control" function of the Unit Supervisor (CRS) affords him/her enough exposure to the MCR panels; therefore, a once per shift solo walkdown is sufficient.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 98 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	2.00
System ID:	1104484
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LGSOPS2005A.3
Topic:	(SRO) Watchstanding Practices - Recall SRO requirement to walkdown MCR panels
Num Field 1:	3.8
Num Field 2:	4.2
Text Field:	2.1.1
Comments:	Level SRO Tier 3 Group N/A KA # and Rating 2.1.1 (3.8/4.2) KA Statement 2.1.1 Knowledge of conduct of operations requirements. References OP-AA-103-102, Rev.12 Examinee References None Learning Objective LGSOPS2005A.3 Question source Bank 986305 Question history None Cognitive level Lower 10 CFR 55 41.10, 43.5 Comments None

99 **ID: 1104486** **Points: 1.00**

SRO

Unit 1 is operating at 100% power, with the following:

- Emergency 220 KV Switching has been performed to support Emergent Maintenance on the 105 breaker

WHICH ONE of the following (per OP-AA-108-107-1002) identifies:

(1) the required oversight for maintenance on the 105 breaker, and
(2) the organization responsible for operation of the breaker?

A. (1) Constant coverage
(2) PECO

B. (1) None; MCR notification required
(2) PECO

C. (1) Constant coverage
(2) LGS OPS

D. (1) None; MCR notification required
(2) LGS OPS

Answer: C

Answer Explanation

Refer to OP-AA-108-107-1002, Attachment 1. The examinee is expected to recognize (by review of the E-0001 drawing) that the "105" breaker is a "Start-Up Source" breaker (for the 10 Station Aux Bus). As such, the left-most column of Attachment 1 applies...which lists "Emergent Maintenance" as an 'A' for the required oversight; i.e., "constant coverage". Refer to Attachment 3, which lists the transmission equipment under the responsibility/ownership of LGS . The 105 breaker is on this list, meaning that it is under the responsibility/ownership of LGS.

'C' is correct: (1) Constant coverage (2) LGS OPS. Correct for the reasons described above.

'A' is wrong: (1) Constant coverage; (2) PECO. Part (2) is plausible to the examinee who fails to thoroughly review Attachment 3.

'B' is wrong: (1) None; MCR notification required; (2) PECO. Part (1) is plausible to the examinee who cannot effectively use the E-0001 drawing to determine that the "105" breaker is in fact a "Start-up Source" breaker. Part (2) is plausible for the same reason as that for choice 'A'.

'D' is wrong: (1) None; MCR notification required; (2) LGS OPS. Part (1) is plausible for the same reason as that for choice 'B'.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 99 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	5	
Difficulty:	2.50	
System ID:	1104486	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	LGSOPS2010.30.A.3	
Topic:	(SRO) Apply OP-AA-108-107-1002, TSO Interface	
Num Field 1:	2.6	
Num Field 2:	3.8	
Text Field:	2.2.17	
Comments:	Level	SRO
	Tier	3
	Group	N/A
	KA # and Rating	2.2.17 (2.6/3.8)
	KA Statement	Knowledge of the process for managing maintenance activities during | power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.
	References	OP-AA-108-107-1002, Rev.7 Electrical Drawing E-0001, Sheet 1, Rev.29
	Examinee References	OP-AA-108-107-1002 E-0001, Sheet 1 (Switchyard portion only)
	Learning Objective	LGSOPS2010.30.A.3
	Question source	Bank 986905, which is a MODIFIED version of Question #96 from the LGS 2012 ILT NRC Exam...Modified as follows: Changed the Switchyard breaker in the stem conditions (from what was the "515" breaker, to now the "105" breaker), which results in changing both parts of the two-part answer.
	Question history	Pre-modified version appeared on LGS 2012 ILT NRC Exam
	Cognitive level	Higher
	10 CFR 55	43.5
	Comments	None

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

100 **ID: 1104488** **Points: 1.00**

SRO

Both Units are operating at 100% power when the MCR is evacuated due to a fire.

25 minutes later:

- Operators establish control of the Units from the Remote Shutdown Panels
- Fire Brigade estinguishes the MCR fire and reports significant damage to MCR equipment

WHICH ONE of the following identifies the **HIGHEST** level of emergency classification for this event?

A. Site Area Emergency - for the fire

B. Site Area Emergency - for the MCR evacuation

C. Alert - for the fire

D. Alert - for the MCR evacuation

Answer: B

Answer Explanation

Refer to EP-AA-1008, LGS EAL Hot Matrix. The MCR fire (i.e., a fire in the Control Enclosure) with damage to MCR equipment meets the ALERT threshold of EAL HA3. The fact that it took longer than 15 minutes for operators to establish plant control from the RSPs meets the SITE AREA EMERGENCY threshold of EAL HS2.

'B' is correct: Site Area Emergency - for the MCR evacuation. Correct for the reasons described above.

'A' is wrong: Site Area Emergency - for the fire. There is no SAE level of EAL for a fire; plausible to the examinee who is distracted more by the fact that the fire not only took longer than 15 minutes extinguish but was also one that causes significant damage to the MCR.

'C' is wrong: Alert - for the fire. Plausible to the examinee who doesn't recognize the significance of the fact that it took 25 minutes to gain plant control but who does recall that the fire meets the threshold for EAL HA3.

'D' is wrong: Alert - for the MCR evacuation. Plausible to the examinee who recalls that the MCR evacuation, alone, is an ALERT per EAL HA2, but doesn't recognize the significance of the fact that it took 25 minutes to gain plant control afterwards.

EXAMINATION ANSWER KEY

LGS 2015 ILT NRC EXAM - SRO

Question 100 Info		
Question Type:	Multiple Choice	
Status:	Active	
Always select on test?	No	
Authorized for practice?	No	
Points:	1.00	
Time to Complete:	3	
Difficulty:	2.50	
System ID:	1104488	
User-Defined ID:	REV 00, 11/17/14	
Cross Reference Number:	NONE	
Topic:	(SRO) Determine HIGHEST EAL for combined MCR fire and evacuation	
Num Field 1:	2.9	
Num Field 2:	4.6	
Text Field:	2.4.41	
Comments:	Level	SRO
	Tier	3
	Group	N/A
	KA # and Rating	2.4.41 (2.9/4.6)
	KA Statement	2.4.41 Knowledge of the emergency action level thresholds and classifications.
	References	EP-AA-1008, Rev.26
	Examinee References	None
	Learning Objective	no specified objective
	Question source	New
	Question history	None
	Cognitive level	Lower
	10 CFR 55	43.5
	Comments	None

EXAMINATION ANSWER KEY
(ANSWERS ONLY)

LGS 2015 ILT NRC EXAM - SRO

1 **ID: 1098269** **Points: 1.00**

Answer: D

2 **ID: 1098270** **Points: 1.00**

Answer: A

3 **ID: 1098287** **Points: 1.00**

Answer: C

4 **ID: 1098291** **Points: 1.00**

Answer: D

5 **ID: 1118907** **Points: 1.00**

Answer: B

6 **ID: 1098294** **Points: 1.00**

Answer: B

7 **ID: 1098325** **Points: 1.00**

Answer: D

8 **ID: 1098327** **Points: 1.00**

Answer: A

9 **ID: 1102907** **Points: 1.00**

Answer: D

10 **ID: 1098329** **Points: 1.00**

Answer: C

11 **ID: 1098330** **Points: 1.00**

Answer: B

EXAMINATION ANSWER KEY
(ANSWERS ONLY)

LGS 2015 ILT NRC EXAM - SRO

12 **ID: 1098331** **Points: 1.00**

Answer: A

13 **ID: 1098336** **Points: 1.00**

Answer: C

14 **ID: 1098344** **Points: 1.00**

Answer: D

15 **ID: 1098353** **Points: 1.00**

Answer: B

16 **ID: 1098424** **Points: 1.00**

Answer: C

17 **ID: 1098425** **Points: 1.00**

Answer: C

18 **ID: 1098446** **Points: 1.00**

Answer: C

19 **ID: 1098470** **Points: 1.00**

Answer: ~~B~~ Ro 3/10/15 Accept A or B

20 **ID: 1098495** **Points: 1.00**

Answer: D

21 **ID: 1098544** **Points: 1.00**

Answer: D

EXAMINATION ANSWER KEY
(ANSWERS ONLY)

LGS 2015 ILT NRC EXAM - SRO

22 **ID: 1098605** **Points: 1.00**

Answer: B

23 **ID: 1098611** **Points: 1.00**

Answer: B

24 **ID: 1109928** **Points: 1.00**

Answer: B

25 **ID: 1110346** **Points: 1.00**

Answer: C

26 **ID: 1102330** **Points: 1.00**

~~Answer: A~~ Ao 3/10/15 Delete Q26

27 **ID: 1102331** **Points: 1.00**

Answer: D

28 **ID: 1102409** **Points: 1.00**

Answer: ~~B~~ Ao 3/10/15 Accept A or B

29 **ID: 1102424** **Points: 1.00**

Answer: B

30 **ID: 1102426** **Points: 1.00**

Answer: C

31 **ID: 1102445** **Points: 1.00**

Answer: C

32 **ID: 1117935** **Points: 1.00**

Answer: A

EXAMINATION ANSWER KEY
(ANSWERS ONLY)

LGS 2015 ILT NRC EXAM - SRO

33 **ID: 1102449** **Points: 1.00**

Answer: D

34 **ID: 1102450** **Points: 1.00**

Answer: A

35 **ID: 1102542** **Points: 1.00**

Answer: C

36 **ID: 1102543** **Points: 1.00**

Answer: A

37 **ID: 1102564** **Points: 1.00**

Answer: B

38 **ID: 1102594** **Points: 1.00**

Answer: B

39 **ID: 1102599** **Points: 1.00**

Answer: A

40 **ID: 1102663** **Points: 1.00**

Answer: D

41 **ID: 1102684** **Points: 1.00**

Answer: A

42 **ID: 1102685** **Points: 1.00**

Answer: B

EXAMINATION ANSWER KEY
(ANSWERS ONLY)

LGS 2015 ILT NRC EXAM - SRO

43 **ID: 1102687** **Points: 1.00**

Answer: C

44 **ID: 1102692** **Points: 1.00**

Answer: A

45 **ID: 1104186** **Points: 1.00**

Answer: D

46 **ID: 1102745** **Points: 1.00**

Answer: B

47 **ID: 1102751** **Points: 1.00**

Answer: C

48 **ID: 1102752** **Points: 1.00**

Answer: A

49 **ID: 1116786** **Points: 1.00**

Answer: D

50 **ID: 1102765** **Points: 1.00**

Answer: A

51 **ID: 1118326** **Points: 1.00**

Answer: B

52 **ID: 1103365** **Points: 1.00**

Answer: A

53 **ID: 1103385** **Points: 1.00**

Answer: D

EXAMINATION ANSWER KEY
(ANSWERS ONLY)

LGS 2015 ILT NRC EXAM - SRO

54 **ID: 1103389** **Points: 1.00**

Answer: B

55 **ID: 1103390** **Points: 1.00**

Answer: B

56 **ID: 1104787** **Points: 1.00**

Answer: D

57 **ID: 1103428** **Points: 1.00**

Answer: A

58 **ID: 1118351** **Points: 1.00**

Answer: C

59 **ID: 1103430** **Points: 1.00**

Answer: C

60 **ID: 1118410** **Points: 1.00**

Answer: C

61 **ID: 1103474** **Points: 1.00**

Answer: D

62 **ID: 1103788** **Points: 1.00**

Answer: D

63 **ID: 1103809** **Points: 1.00**

Answer: D

EXAMINATION ANSWER KEY
(ANSWERS ONLY)

LGS 2015 ILT NRC EXAM - SRO

64 **ID: 1103884** **Points: 1.00**

Answer: A

65 **ID: 1103891** **Points: 1.00**

Answer: D

66 **ID: 1103894** **Points: 1.00**

Answer: B

67 **ID: 1103944** **Points: 1.00**

Answer: D

68 **ID: 1103946** **Points: 1.00**

Answer: C

69 **ID: 1103947** **Points: 1.00**

Answer: C

70 **ID: 1103964** **Points: 1.00**

Answer: A

71 **ID: 1103966** **Points: 1.00**

Answer: C

72 **ID: 1104004** **Points: 1.00**

Answer: A

73 **ID: 1104005** **Points: 1.00**

Answer: B

74 **ID: 1104006** **Points: 1.00**

Answer: D

EXAMINATION ANSWER KEY
(ANSWERS ONLY)

LGS 2015 ILT NRC EXAM - SRO

75 **ID: 1104526** **Points: 1.00**

Answer: A

76 **ID: 1102804** **Points: 1.00**

Answer: A

77 **ID: 1118645** **Points: 1.00**

Answer: A

78 **ID: 1102809** **Points: 1.00**

Answer: C

79 **ID: 1102824** **Points: 1.00**

Answer: C

80 **ID: 1102826** **Points: 1.00**

Answer: A

81 **ID: 1104650** **Points: 1.00**

Answer: ~~C~~ 3/10/15 Accept C or D

82 **ID: 1102864** **Points: 1.00**

Answer: D

83 **ID: 1118706** **Points: 1.00**

Answer: B

84 **ID: 1102904** **Points: 1.00**

Answer: B

EXAMINATION ANSWER KEY
(ANSWERS ONLY)

LGS 2015 ILT NRC EXAM - SRO

85 **ID: 1102906** **Points: 1.00**

Answer: D

86 **ID: 1104204** **Points: 1.00**

Answer: A

87 **ID: 1104205** **Points: 1.00**

Answer: D

88 **ID: 1104206** **Points: 1.00**

Answer: C

89 **ID: 1104267** **Points: 1.00**

Answer: D

90 **ID: 1104304** **Points: 1.00**

Answer: A

91 **ID: 1104380** **Points: 1.00**

Answer: A

92 **ID: 1118691** **Points: 1.00**

Answer: D

93 **ID: 1104389** **Points: 1.00**

~~Answer: B~~ Delete 093
3/10/15

94 **ID: 1104465** **Points: 1.00**

Answer: C

95 **ID: 1118715** **Points: 1.00**

Answer: B

EXAMINATION ANSWER KEY
(ANSWERS ONLY)

LGS 2015 ILT NRC EXAM - SRO

96 **ID: 1104467** **Points: 1.00**

Answer: D

97 **ID: 1104468** **Points: 1.00**

Answer: C

98 **ID: 1104484** **Points: 1.00**

Answer: B

99 **ID: 1104486** **Points: 1.00**

Answer: C

100 **ID: 1104488** **Points: 1.00**

Answer: B