1	ID: 1740800	Points: 1.00

Unit 2 is operating at rated power when the "2C" SRV inadvertently opens.

What indications are observed on the 20C603 panel for Total Main Steam Flow indication and how do Turbine Control Valves (TCVs) respond?

	Indicated Total Steam Flow	Total Control Valve Position
A.	Lowers	Closes
B.	Lowers	Opens
C.	Rises	Closes
D.	Rises	Opens

Answer: A

Answer Explanation

See P&ID M-0001 sheet 3 and M-0041 sheet 5, for the physical relationship between the SRVs and the MSL flow
elements; the SRVs are upstream. Thus, an open SRV (discharging to the suppression pool)
robs steam flow away from the MSL and its flow elements (M-0041 sheet 5). Therefore, indicated Total Main
Steam Flow on flow recorder FR 2R607(at 20C603) lowers.
The steam pressure sensed at the pressure averaging manifold (PT-201A and 201B)
also lowers, causing the DEHC system pressure regulator to respond by throttling the
TCVs in the closed direction in an effort to control steam pressure.

Therefore the correct answer is "A"

B, C and D are plausible to the student who does not recall the relative locations of the SRV connections in relation to the flow nozzles or EHC pressure transmitters.

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:	0.00
System ID:	1740800
User-Defined ID:	Q #01 BANK
Lesson Plan Objective:	LGSOPS0001B.2B
Topic:	Unit 2 is operating at rated power when the "2C" SRV inadvertently opens. What indications are o
RO Importance:	3.6
SRO Importance:	3.7
K/A Number:	(239002 K1.04

Comments:	
	General Data
	Level: RO
	Tier: 2
	Group: 1
	KA # and Rating: 239002 SRV K1.04 3.6 / 3.7
	KA Statement: **Knowledge of the physical connections and/or causeeffect relationships between RELIEF/SAFETY VALVES and the following:** Main steam
	Cognitive level: Higher
	Safety Function: 3 - Reactor Pressure Control
	10 CFR 55: 41.2
	Technical Reference with Revision No: M-0001 Sheet3 **Rev #:**
	Justification for Non SRO CFR Link:
	Question History: (i.e. LGS NRC-05, OYS CERT-04): GGNS 2012 NRC exam
	Question Source: (i.e. New, Bank, Modified): Bank
	Low KA Justification (if required):
	Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)
	ILT
	Supplied Ref (If appropriate): (i.e. ABN-##): None
	LORT
	PRA: (i.e. Yes or No or #)
	LORT Question Section: (i.e, A-Systems or B-Procedures)
	Comments
	Grand Gulf Nuclear Station (GGNS) 2012 NRC ILT #4

2	ID: 1740823	Points: 1.00

Unit 2 is operating at 100% power.

WHICH ONE of the following describes the expected effect on the APRM reading(s) and the APRM Gain Adjustment Factors (AGAF) data of a 3-D Monicore Periodic Log (P-1), over the next 30 days due to LPRM aging?

Assume core thermal power will be constant for the next 30 days.

	APRM Reading	AGAF
A.	Increase	Increase
B.	Increase	Decrease
C.	Decrease	Increase
D.	Decrease	Decrease

Answer: C

Answer Explanation

Answer: APRM reading decreases   AGAF increases

The sensitivity of the LPRMs changes with exposure.

As LPRMs age the effect seen on the APRMs is a lower than normal reading. The APRMs are than adjusted per the Gain adjustments to reflect actual real power. APRM Gain Adjustment Factors are used to adjust APRM amplifiers so that APRM reflects real power as calculated by the heat balance.

APRM gain adjustment factors are given by:

$$AGAF(I)=\frac{\dfrac{CTP}{REFCTP}}{\dfrac{RAP(I)}{REFRAP(I)}}=\frac{Actual}{Indicated}$$

From DBD L-S-20

3.3.1.3.3 LPRM Operation

The LPRM is designed to provide incore monitoring during power operation. Each LPRM channel has user adjustable gain. Changing to a higher gain will compensate the loss in sensitivity of the LPRM detector. This gain is adjusted after each Process Computer/TIP core scan. The channel can be manually bypassed if failed. All 172 LPRMs feed the APRMs; anc all 172 LPRMs feed the RBMs, and Unit 1 and Unit 2 PM Computer for thermal limit calculations and facilitation of LPRM/APRM calibration. {6.1.14.9}

<u>BASIS</u>: Changing the gain will prolong the LPRM detector life. Bypassing a LPRM channel will allow th associated APRM, OPRM and RBM channels to remain operable until the minimum input requirement is violated. The design is provided to meet the design inputs of Other Design Inputs {2.5.2}.

<u>Distracters:</u>

<u>APRM reading</u>   <u>AGAF</u>

<u>Increase</u>   <u>Increase</u> - Plausible to the student that believes that the sensitivity of the LPRMs rise via exposure

<u>Increase</u>   <u>Decrease</u> - Plausible to the student that believes that the sensitivity of the LPRMs rise via exposure and plausible to the student that inverts the numerator and denominator to the AGAF equation.

<u>Decrease</u>   <u>Decrease</u> – plausible to the student that inverts the numerator and denominator to the AGAF equation.

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:	0.00
System ID:	1740823
User-Defined ID:	Q #02 BANK
Lesson Plan Objective:	LLOT0038E.2
Topic:	Predict Response of APRM Readings and AGAFs to LPRM Aging
RO Importance:	2.6
SRO Importance:	2.9
K/A Number:	215005 K1.07

Comments:	

General Data			
Level	RO		
Tier	2		
Group	1		
KA # and Rating	215005 K1.07 2.6/2.9		
KA Statement	**Knowledge of the physical connections and/or causeeffect relationships between AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM and the following:** K1.07 Process computer, performance monitoring system		
Cognitive level	higher		
Safety Function	7		
10 CFR 55	41.6 & 7		
Technical Reference with Revision No:	DBD L-S-20	**Rev #:**	
Justification for Non SRO CFR Link:	n/a		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	bank		
Question Source: (i.e. New, Bank, Modified)	bank 561184		
Low KA Justification (if required):	n/a		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

3	ID: 1740930	Points: 1.00

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

- D134-R-E is de-energized

A steam leak develops in the U1 RCIC room with room temperatures reaching 195° F.

WHICH ONE of the following identifies the RCIC Isolation Valve status for the above conditions?

	HV-49-1F007, RCIC MAIN STEAM SUPPLY INBRD PCIV (INBOARD)	HV-49-1F008, RCIC STEAM LINE OUTBOARD PCIV (OUTBOARD)
A.	Open	Open
B.	Open	Closed
C.	Closed	Open
D.	Closed	Closed

Answer: B

Answer Explanation
D134-R-E powers the motor-actuator for the RCIC Inboard Steam Line Isolation Valve, HV-49-1F007. Therefore, without D134-R-E, there is no way to electrically stroke the 1F007 valve. The Outboard Steam Line Isolation, HV-49-1F008, motor-actuator is powered from D114-R-G1 and is, therefore, unaffected by the loss of D134-R-E. The stem of the question provides room temperature of 195° F which is above the RCIC room temperature isolation set point (Tech Spec Table 3.3.2-2 5.d) of 180° F. With an isolation signal, normally both isolations valves will close, due to D134-R-E being de-energized only HV-049-1F008 will close. 'B' is correct for the above reasons. 'A' is wrong but plausible to the student who incorrectly recalls the RCIC room isolation temperature setpoint and believes that the set point has not been reached. Therefore determining that both isolation valves remain open. 'C' is wrong but plausible to the student who incorrectly recalls the power supplies to the isolation valves and believes that HV-049-1F008 is powered from D134-R-E. 'D' is wrong but plausible to the student who incorrectly recalls the power supplies to the isolation valves and believes that neither isolation valve is powered from D134-R-E.

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:	0.00
System ID:	1740930
User-Defined ID:	Q #03 NEW
Lesson Plan Objective:	LGSOPS0049.4
Topic:	Impact of Loss of 480V AC on RCIC isolation valve
RO Importance:	3.4
SRO Importance:	
K/A Number:	217000 K6.01

Comments:	

General Data			
Level	RO		
Tier	2		
Group	1		
KA # and Rating	217000 K6.01 RO Importance 3.4		
KA Statement	217000 Reactor Core Isolation Cooling System (RCIC) K2. Knowledge of electrical power supplies to the following: Motor operated valves		
Cognitive level	high		
Safety Function	2 - Reactor Water Inventory Control		
10 CFR 55	41.7		
Technical Reference with Revision No:	E-0058 Sheet 003	**Rev #:**	042
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	NEW		
Question Source: (i.e. New, Bank, Modified)	NEW		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

4	ID: 1740951	Points: 1.00

WHICH ONE of the following describes the effect of placing the HPCI Emergency Shutdown Switch (HS-56-162) to "OFF" at the Remote Shutdown Panel (RSP)?

A. Shuts down HPCI by removing power to the speed governor which closes the control valve

B. Shuts down HPCI by closing the steam admission valve HV-55-1F001

C. Trips the HPCI turbine by energizing the trip solenoid SV1

D. Isolates HPCI by closing the outboard steam isolation valve HV-55-1F003

Answer: A

Answer Explanation
Per E41-1040 Sht 2 E41-1040 Sht 9 taking HS-56-162 to OFF removes power to the speed governor. This keylock functions regardless of the position of the Remote Shutdown Panel Transfer Switches. Per lesson plan LGSOPS0055 page 21 F. Manual Shutdown from the Remote Shutdown Panel 1. Operating the HPCI EMERG S/D Switch,HSS-56-*62, isolates control power to the Governor resulting in the Turbine Control Valve failing close. A correct for the above reasons B Incorrect, plausible to the candidate incorrectly recalls which valve HS-56-162 closes C Incorrect, plausible to the candidate who equates HS-56-162 in off to tripping the HPCI turbine D incorrect, plausible to the candidate who believe that HS-56-162 performs the same function at the RSP as as the HPCI Isolation PB in the MCR

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:	0.00
System ID:	1740951
User-Defined ID:	Q #04 BANK
Lesson Plan Objective:	LOT-0340, OBJ. #3
Topic:	describes the operation of placing the Emergency Shutdown Switch (HS-56-162) to "OFF" at the Remot
RO Importance:	2.5
SRO Importance:	2.7
K/A Number:	206000 K2.04

Comments:

General Data			
Level	RO		
Tier	2		
Group	1		
KA # and Rating	206000 K2.04 2.5/2.7		
KA Statement	**High Pressure Coolant Injection System Knowledge of electrical power supplies to the following:** Turbine control circuits: BWR-2,3,4		
Cognitive level	lower		
Safety Function	2		
10 CFR 55	41.7		
Technical Reference with Revision No:	E41-1040 Sht 2 E41-1040 Sht 9	**Rev #:**	18 19
Justification for Non SRO CFR Link:	n/a		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	bank 557618		
Question Source: (i.e. New, Bank, Modified)	bank		
Low KA Justification (if required):	n/a		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)			
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

5	ID: 1741393	Points: 1.00

Unit 2 is in a refueling outage with the RPV Head fully tensioned, with the following:

- 2B RHR is in Shutdown Cooling (SDC)
- SDC flow rate is 9000 gpm
- RWCU is in service
- RWCU bottom head drain temperature is reading 260°F
- 2B RHR heat exchanger inlet temperature is 145°F
- 2B heat exchanger outlet temperature is 125°F
- 2B recirc suction and discharge valves are open

WHICH ONE of the following identifies (1) reactor temperature and (2) reactor pressure?

A. (1) 125°F
(2) 0 psig

B. (1) 145°F
(2) 0 psig

C. (1) 260°F
(2) 20.7 psig

D. (1) 260°F
(2) 35.4 psig

Answer: C

Answer Explanation

Correct answer: 'C'. With 'B' loop SDC in service and both the 2B recirc suction and discharge valves open the shutdown cooling flow is bypassing the reactor. Rather than the cooled water returning to the reactor vessel through the Jet Pumps, the cooled water is returning to the suction of the RHR pump through the recirc pump. With bypass flow, RHR temperature indication is not valid. Therefore, the operator must determine that with RWCU in service RPV Bottom Head temperature is an accurate temperature reading. Based on information given in the stem Reactor temperature is 260°F. Using the steam tables and a temperature of 260°F the student is to determine that the pressure of the saturated system is 20.7 psig.

125° is not correct because RHR hx outlet temp is not reflective of core exit temp.

145° is not correct because cooling flow is bypassing the reactor.

260° 35.4 psig is incorrect plausible if candidate misreads the steam tables and uses the psiA value.

From S51.8.B:

3.13 Operating RHR in the Shutdown Cooling Mode
AND Recirc in that loop is out of service with both Recirculation Pump valves HV-43-*F023A(B), SUCTION
AND HV-43-*F031A(B), DISCHARGE, open will create a Shutdown Cooling Bypass Leakage Path
AND could result in an unexpected OPCON Mode change due to decreased Shutdown Cooling capability.

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:	0.00
System ID:	1741393
User-Defined ID:	Q #05 NEW
Lesson Plan Objective:	LGSOPS0051.13D
Topic:	Determine RCS temperature given SDC configuration
RO Importance:	3.3
SRO Importance:	3.6
K/A Number:	205000 K3.01

<table>
<tr><td>Comments:</td><td>

General Data			
Level	RO		
Tier	2		
Group	1		
KA # and Rating	205000 K3.01 3.3/3.6		
KA Statement	**Shutdown Cooling System (RHR Shutdown Cooling Mode) Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following:** Reactor pressure		
Cognitive level	higher		
Safety Function	4		
10 CFR 55	41.7, 41.5		
Technical Reference with Revision No:	S51.8.B page 2 M-0043 sht 2&3	**Rev #:**	81
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	new		
Question Source: (i.e. New, Bank, Modified)	new		
Low KA Justification (if required):	n/a		
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)	new		
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	Steam Tables		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

S51.8.B, page 2
Steam Tables</td></tr>
</table>

6	ID: 1741408	Points: 1.00

Unit 2 is operating at 100% power:

- D24 DG is paralleled with the 201 Safeguard Bus
- An inadvertent LOCA signal is received on the 2D Core Spray Subsystem

WHICH ONE of the following identifies the impact on the D24 DG Output Breaker and MCR Speed Governor and Voltage Regulator controls, if any?

	D24 Output Breaker	MCR DG Speed/Voltage Controls
A.	Trips	Still function
B.	Trips	Will NOT function
C.	Remains closed	Still function
D.	Remains closed	Will NOT function

Answer: B

Answer Explanation

A Incorrect, plausible to the candidate that mistakenly believes the EDG will remain in Droop mode (with speed and voltage controls available in the MCR).

B Correct, With the DG operating in parallel with an offsite source, a LOCA signal will trip the output breaker and convert the DG from Droop to Isochronous mode. The LOCA signal also results in disabling the speed governor and voltage regulator controls with the governor locked into a 60 Hz frequency and the voltage regulator operating at a fixed 4280 VAC.

C Incorrect, Plausible to the candidate that believes that since the EDG is running in parallel with the D24 BUS, the EDG output breaker will remain closed. This would be true for a LOOP situation but not a LOCA signal and plausible to the candidate that mistakenly believes the EDG will remain in Droop mode (with speed and voltage controls available in the MCR).

D Incorrect, Plausible to the candidate that believes that since the EDG is running in parallel with the D24 BUS, the EDG output breaker will remain closed. This would be true for a LOOP situation but not a LOCA signal.

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:	0.00
System ID:	1741408
User-Defined ID:	Q #06 BANK MOD
Lesson Plan Objective:	LGSOPS0092B.06
Topic:	Effect On Breaker And Speed/Voltage Controls From LOCA While Synchronized
RO Importance:	2.9
SRO Importance:	3.0
K/A Number:	209001 K3.03

Comments:	

General Data	
Level	RO
Tier	2
Group	1
KA # and Rating	209001 K3.03 2.9/3.0
KA Statement	**Low Pressure Core Spray System Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on following:**Emergency generators
Cognitive level	Lower
Safety Function	2
10 CFR 55	41.7, 41.8
Technical Reference with Revision No:	**Rev #:**
Justification for Non SRO CFR Link:	n/a
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Bank 846362 slightly modified
Question Source: (i.e. New, Bank, Modified)	Bank 846362 slightly modified
Low KA Justification (if required):	n/a
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	Modified stem to be a C.S logic failure to cause DG start rather than actual LOCA. Change to match KA
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	
LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

7	ID: 1741563	Points: 1.00

Unit 1 plant conditions are as follows:

- Reactor power is 31%
- Reactor level is +35"
- FWLC is as follows:
  - 1A RFP in Manual at $1.5X10^6$ pounds per hour
  - 1B RFP in Auto at $2.8X10^6$ pounds per hour
  - 1C RFP secured

The 1B RFP trips

WHICH ONE of the following identifies the Recirc Pump speed runback signal, and the reason for the signal?

	Recirc Runback Received	Reason for Recirc Runback
A.	28% speed	Ensure NPSH to Recirc Pumps & Jet Pumps.
B.	28% Speed	Prevent RFP trip on low suction pressure
C.	42% speed	Ensure NPSH to Recirc Pumps & Jet Pumps
D.	42% Speed	Prevent RFP trip on low suction pressure

Answer: A

Answer Explanation

Automatic RRP Runbacks
1) High Limit Runback - 699 rpm (42%)
a) >12Mlbm/hr FW Flow AND any condensate pump breaker tripped.
b) < 27.5" Rx Level AND any FW loop flow less than 0.94 Mlbm/hr flow.
2) Low Limit Runback - 466 rpm (28%)
a) < +12.5" Rx Level
b) Discharge valve not full open
c) FW Flow less than 2.8 Mlbm/hr
3) Speed Hold has no effect on any Automatic Runback.
4) Runback Reset pushbutton resets all runback signals for both Recirc Pumps

A Correct, 28% runback on <$2.8x10^6$ lbm/hr feedwater flow is to ensure NPSH to Recirc and jet pumps

B Incorrect, 28% runback is correct but the basis, prevent RFP trip is basis for 42% runback, plausible to the candidate the mistakenly recalls that basis for a different Recirc Runback.

C Incorrect, <$2.8x10^6$ feedwater flow is a 28% runback. Plausible to the candidate who mistakenly recalls the recirc speed runback for a different event.

D Incorrect, <$2.8x10^6$ feedwater flow is a 28% runback prevent RFP trip is basis for 42% runback. Plausible to the candidate who mistakenly recalls the recirc speed runback for a different event and plausible to the candidate the mistakenly recalls that basis for a different Recirc Runback.

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:	0.00
System ID:	1741563
User-Defined ID:	Q #07 NEW
Lesson Plan Objective:	LGSOPS0043A.IL7 & LLOT0540.4A
Topic:	DFLCS - reason for total flow less than 2.8 Mlbm RR Runback
RO Importance:	3.0
SRO Importance:	3.1
K/A Number:	295002 K4.01

Comments:	

General Data	
Level	RO
Tier	2
Group	1
KA # and Rating	295002 K4.01 3.0/3.1
KA Statement	**Reactor Water Level Control System Knowledge of REACTOR WATER LEVEL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following:** Ensuring adequate NPSH for recirculation pumps: Plant-Specific
Cognitive level	High
Safety Function	2 **Reactor Water Inventory Control**
10 CFR 55	41.7
Technical Reference with Revision No:	**Rev #:**
Justification for Non SRO CFR Link:	n/a
Question History: (i.e. LGS NRC-05, OYS CERT-04)	new
Question Source: (i.e. New, Bank, Modified)	
Low KA Justification (if required):	na
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	None
LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

8	ID: 1741662	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'.

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:	0.00
System ID:	1741662
User-Defined ID:	Q #08 BANK
Lesson Plan Objective:	LLOT0071.7
Topic:	Recall IRM Rod Blocks
RO Importance:	3.7
SRO Importance:	3.7
K/A Number:	215003 K 4.01

Comments:	

General Data			
Level	RO		
Tier	2		
Group	1		
KA # and Rating	215003 K4.01 3.7/3.7		
KA Statement	**Intermediate Range Monitor (IRM) System Knowledge of INTERMEDIATE RANGE MONITOR (IRM) SYSTEM design feature(s) and/or interlocks which provide for the following:** Rod withdrawal blocks		
Cognitive level	lower		
Safety Function	7 Instrumentation		
10 CFR 55	41.6,41.7		
Technical Reference with Revision No:	ARC-MCR-107, F3, ARC-MCR-107, H3,	**Rev #:**	Rev.1Rev.0
Justification for Non SRO CFR Link:			
Question History: (i.e. LGS NRC-05, OYS CERT-04)	bank 1149956 which is 560453 modified		
Question Source: (i.e. New, Bank, Modified)	bank 1149956		
Low KA Justification (if required):			
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

9	ID: 1741722	Points: 1.00

If not properly ventilated, an accumulation of ___(1)___ in the battery room would occur. This accumulation would occur faster during battery ___(2)___

	(1)	(2)
A.	hydrogen	charging
B.	hydrogen	discharging
C.	sulfuric acid	charging
D.	sulfuric acid	discharging

Answer: A

Answer Explanation

From Lesson plan LGSOPS0078

- Hydrogen production is greater during charging. Sulfuric acid, although produced during charging will remain in aqueous form

A correct for the reasons stated above

B Incorrect Hydrogen production is greater during charging. Plausible to the examinee who believe that the increased load on the battery during discharge will increase hydrogen production

C incorrect Sulfuric acid, although produced during charging will remain in an aqueous form and not buildup in the battery room. Plausible to the examinee who does not fully understand battery charging.

D incorrect Sulfuric acid, although produced during charging will remain aqueous form and not buildup in the battery room plausible for the reasons stated above

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:	0.00
System ID:	1741722
User-Defined ID:	Q #09 NEW
Lesson Plan Objective:	LGSOPS0078.1C
Topic:	Battery H2 buildup
RO Importance:	2.6
SRO Importance:	2.9
K/A Number:	263000 K5.01

Comments:	General Data			
	Level	RO		
	Tier	2		
	Group	1		
	KA # and Rating	263000 K5.01 2.6/2.9		
	KA Statement	**D.C. Electrical Distribution Knowledge of the operational implications of the following concepts as they apply to D.C. ELECTRICAL DISTRIBUTION :** Hydrogen generation during battery charging		
	Cognitive level	lower		
	Safety Function	6		
	10 CFR 55	41.5		
	Technical Reference with Revision No:	M-078 sht 4	**Rev #:**	22
	Justification for Non SRO CFR Link:	n/a		
	Question History: (i.e. LGS NRC-05, OYS CERT-04)	new		
	Question Source: (i.e. New, Bank, Modified)	new		
	Low KA Justification (if required):	n/a		
	Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)	new		
	ILT			
	Supplied Ref (If appropriate): (i.e. ABN-##)	none		
	LORT			
	PRA: (i.e. Yes or No or #)			
	LORT Question Section: (i.e, A-Systems or B-Procedures)			
	Comments			

10	ID: 2029729	Points: 1.00

Unit 2 plant conditions are as follows:

- GP-2, Normal Plant Startup, is in progress
- The Neutron Monitoring Overlap Surveillance Test is complete

SRMs are being retracted with the following conditions:

IRM CHANNEL	IRM RANGE
A	4
B	3
C	2
D	4
E	BYPASSED
F	3
G	5
H	3

While being retracted, the reading from the "2A" SRM drops to 90 cps.

WHICH ONE of the following describes the status of the SRM RETRACTED WHEN NOT PERMITTED alarm and the rod block response?

	SRM RETRACTED WHEN NOT PERMITTED ALARM	ROD BLOCK
A.	off	enforced
B.	off	NOT enforced
C.	on	enforced
D.	on	NOT enforced

Answer: C

Answer Explanation

Answer: SRM retract permit is on: Rod Block is enforced

<u>CONTROL ROD WITHDRAWAL BLOCKS</u>, through interface to Reactor Manual Control System (RMCS). The SRMs produce a control rod withdrawal block under the following conditions:

- UPSCALE - channel output greater than 1 x 10^5 cps
- INOP – Channel Inoperative is caused by any one of the following:

Low detector high voltage (< 90% of actual high voltage value)
Module Unplugged
Drawer selector switch (located in Aux Equipment Room) not in operate

- DOWNSCALE – channel output less than 3 cps
- SRM not fully inserted and less than 100 cps. (This prevents premature withdrawal of the detectors).

<u>CONTROL ROD WITHDRAWAL BLOCK Bypass Conditions</u>

- Manual Joystick located on Panel *0C603 can manually bypass one SRM channel at a time.
- All SRM Rod Blocks are bypassed when the Reactor Mode Switch is in RUN
- All SRM Rod Blocks are bypassed when all 4 associated IRM channels are on Range 8 or above (or bypassed)

IRM Channels A, C, E, and G for SRM Channels A and C
IRM Channels B, D, F and H for SRM Channels B and D
The SRM Upscale Rod Block is bypassed under ANY of the following conditions:

- Reactor Mode Switch in RUN
- All 4 associated IRM channels on Range 8 or above (or bypassed)

SRM Downscale Rod Block is bypassed when all 4 associated IRM Channels are on Range 3 or above (or bypassed)
The SRM Rod Block on Detector Not Fully Inserted is bypassed on EITHER of the following conditions:

- SRM Channel count rate > 100 cps
- All 4 associated IRM Channels are on Range 3 or above (or bypassed)
- The SRM Upscale Hi-Hi Trip is bypassed when the RPS shorting links are installed

A Wrong - plausible to the candidate that fails to recall the associated IRMs that provide interlocks to the 2A SRM (IRMs A,C,E,G Vice IRMs B,D,F,H).
B Wrong - plausible to the candidate that fails to recall the associated IRMs that provide interlocks to the 2A SRM (IRMs A,C,E,G Vice IRMs B,D,F,H) and plausible to the candidate that fails to recall the SRM count rate that causes the rod out block
C Correct for the above reasons
D Wrong - plausible to the candidate that fails to recall the SRM count rate that causes

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:	0.00
System ID:	2029729
User-Defined ID:	Q #10
Lesson Plan Objective:	LGSOPS0074.2
Topic:	SRM Operation
RO Importance:	2.6
SRO Importance:	2.6
K/A Number:	215004 K5.01

Comments:

General Data	
Level	RO
Tier	2
Group	1
KA # and Rating	215004 K5.01 2.6/2.6
KA Statement	**Source Range Monitor (SRM) System Knowledge of the operational implications of the following concepts as they apply to SOURCE RANGE MONITOR (SRM) SYSTEM :** Detector operation
Cognitive level	high
Safety Function	7 instrumentation
10 CFR 55	41.6,41.7
Technical Reference with Revision No:	ARC-MCR-207 I4 — **Rev #:** 01
Justification for Non SRO CFR Link:	N/A
Question History: (i.e. LGS NRC-05, OYS CERT-04)	2015 CERT
Question Source: (i.e. New, Bank, Modified)	Bank 1149961
Low KA Justification (if required):	N/A
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	N/A
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	None
LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

11	ID: 1742911	Points: 1.00

Unit 2 is operating at 20% power during power ascension with the main generator synched to the grid. House Loads have been transferred to the Aux Busses per S91.6.A, TRANSFERRING HOUSE LOADS TO UNIT AUXILIARY TRANSFORMER

A Main Turbine trip / Generator lockout occurs.

WHICH ONE of the following describes the status of the Startup Source and the 13.2 KV ASD Supply Breakers following the Generator Lockout?

	10-22 and 20-21 Breakers	13.2 KV ASD Breakers
A.	Closed	Closed
B.	Closed	Open
C.	Open	Closed
D.	Open	Open

Answer: B

Answer Explanation

A Closed Closed is incorrect as the 13.2 KV ASD Breakers trip on a Aux Bus Fast transfer. Plausible to the candidate who confuses the Fast Transfer trip with the RPT EOC trip which would not actuate at 20% power.

B Closed Open - Correct

C Open Closed is incorrect as the Startup Breakers should be closed following a fast transfer and the 13.2 KV ASD breakers would be Open. Plausible to the candidate who does not remeber the fast transfer action of the Generator Trip or the Auto Tripping action of the 13 KV Breakers on a Generator Trip.

D Open Open is incorrect since the startup breakers would be closed. Plausible to the candidate who does not recall the fast transfer action on a Generator Trip.

Actuating any one of the unit protection lockout relays will initiate the following actions:

(1) Trip of the Main Generator output breakers - Isolates fault to or from the main generator

(2) Trip of the Alterrex exciter field breaker - Limits internal damage to the Alterrex exciter

(3) Trip of the Main Turbine - Prevents main turbine overspeed due to loss of load

(4) Trip of the Stator Water Cooling pumps - Limits water input to the main generator stator if a stator bar has ruptured

(5) Auxiliary Bus fast transfer – 11, 12 or 21, 22 Busses transfer from the Main Generator to the Start-Up busses - Maintains power to selected vital loads

(6) Transfer of the main generator voltage regulator to MANUAL - Prevents voltage regulator damage

(7) Trip of the Reactor Recirculation pumps, if powered from the main generator - Minimizes reactor power during the transient

(8) Energization of the main generator output breaker failure circuit - To ensure main generator has been isolated from the 220 KV switchyard

(9) Trip of the main unit transformer cooling units - No need for cooling of main unit transformer, and/or to protect against a fault in the main unit transformer

(10)Trip of the unit auxiliary transformer cooling unit - No need for cooling of unit auxiliary transformer, and/or to protect against a fault in the unit auxiliary transformer

Question 11 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1742911
User-Defined ID:	Q #11
Lesson Plan Objective:	LGSOPS0035.6
Topic:	Electrical Distribution Main Generator Trip
RO Importance:	3.5
SRO Importance:	3.7
K/A Number:	262001 K6.03

Comments:	General Data	
	Level	RO
	Tier	2
	Group	1
	KA # and Rating	262001 K6.03 3.5 / 3.7
	KA Statement	**Knowledge of the effect that a loss or malfunction of the following will have on the A.C. ELECTRICAL DISTRIBUTION:** Generator trip
	Cognitive level	High
	Safety Function	6
	10 CFR 55	41.7
	Technical Reference with Revision No:	S91.6.A E-0001 — **Rev #:** 1530
	Justification for Non SRO CFR Link:	
	Question History: (i.e. LGS NRC-05, OYS CERT-04)	
	Question Source: (i.e. New, Bank, Modified)	New
	Low KA Justification (if required):	
	Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	
	ILT	
	Supplied Ref (If appropriate): (i.e. ABN-##)	None
	LORT	
	PRA: (i.e. Yes or No or #)	
	LORT Question Section: (i.e, A-Systems or B-Procedures)	
	Comments	

12	ID: 1798989	Points: 1.00

Unit 1 is operating at 100% Power when the following process radiation monitors momentarily spike to the indicated values due to an electrical transient:

RISH-26-1K609C, REACTOR BLDG VENTILATION MON: 1.4 mR/hr

and

RISH-26-1K609D, REACTOR BLDG VENTILATION MON: 1.7 mR/hr

WHICH ONE of the following identifies the status of the Standby Gas Treatment System (SGTS) for the above conditions:

	A SGTS Fan	B SGTS Fan
A.	NOT Running	NOT Running
B.	NOT Running	Running
C.	Running	NOT Running
D.	Running	Running

Answer: B

Answer Explanation

From the stem the candidate determines that both process radiation monitors malfunctions took them above the isolation set point of 1.35 mR/hr. Based on this information the candidate concludes that the B SGTS fan is now in service.

From LGSOPS0076 Lesson Plan:

REACTOR ENCLOSURE ISOLATION SIGNALS			
SIGNAL	DIVISION 1	DIVISION 2	SETPOINT
MANUAL	HS76-*78A	HS76-*78B	Arm & Depress
EXH. HI RAD	A and B Inst.	C and D Inst.	1.35mR/Hr
LOW RPV LEVEL/HIGH DW PRESSURE	A and B Inst.	C and D Inst.	-38",1.68#
SGTS DAMPER OPEN	HV76-*96	HV76-*97	Not full closed
LOW ZONE DP	A	B	-0.1"H_2O for 50 minutes (still a vacuum, but not enough vacuum)
REFUEL FLOOR ISOLATION	Any Div 1 Isol.	Any Div 2 Isol.	*

A Wrong - Plausible to the candidate that recalls the incorrect isolation set point (2.0 mR/hr for refuel ventilation exhaust).

B Correct for the above reasons

C Wrong - Plausible to the candidate that recalls the incorrect Logic system association (i.e. they incorrectly recall that the 0A Fan is associated with an Div 2 isolation)

D Wrong - plausible to the candidate the confuses the isolation logic with that of the CREFAS system where an upscale condition in the C detector would start the A Fan and an upscale on the D detector would start the B Fan.

Question 12 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1798989
User-Defined ID:	Q #12
Lesson Plan Objective:	LGSOPS0076.05
Topic:	SGTS - Response to Process Rad Monitor Spikes
RO Importance:	2.9
SRO Importance:	
K/A Number:	261000 K6.04

Comments:

General Data			
Level	RO		
Tier	2		
Group	1		
KA # and Rating	261000 K6.04 RO Importance 2.9		
KA Statement	261000 SGTS K6.04 - Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM : Process radiation monitoring		
Cognitive level	High		
Safety Function	9- Radioactivity Release		
10 CFR 55	41.7		
Technical Reference with Revision No:	LGSOPS0076	**Rev #:**	0 0 0
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	N/A		
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

13	ID: 1744279	Points: 1.00

The 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 one of the PRO actions and the reason why the action is performed?

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

A.	200-300 KW.	Prevent oil buildup in the exhaust manifold
B.	200-300 KW.	Prevent motoring the generator
C.	100-150 KVAR.	Prevent oil buildup in the exhaust manifold
D.	100-150 KVAR.	Prevent motoring the generator

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: 200-300 KW. Plausible to the examinee who recalls a low load will cause oil buildup in the exhaust manifold, but who confuses "200-300" band KW requirement for preventing a reverse power trip with the 855KW for oil buildup.

'C' is wrong: 100-150 KVAR. Plausible to the examinee who properly recalls both values from the ST (real load value and reactive load value), but who incorrectly associates the KVAR requirement with the 855KW for oil buildup.

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

Question 13 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1744279
User-Defined ID:	Q #13 BANK
Lesson Plan Objective:	LGSOPS0092B.IL6
Topic:	Recall how the operator prevents a reverse power D/G trip when paralleling
RO Importance:	3.0
SRO Importance:	3.1
K/A Number:	264000 A1.09

Comments:	

General Data			
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)		
Cognitive level	Low		
Safety Function	6 - Electrical		
10 CFR 55	41.5		
Technical Reference with Revision No:	ST-6-092-312-1	**Rev #:**	96
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	2015 CERT Bank 1149941		
Question Source: (i.e. New, Bank, Modified)	Bank 1149941		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	N/A		
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

14	**ID: 1744293**	**Points: 1.00**

Unit 1 reactor pressure is 1045 psig when a T-112, Emergency Blowdown is directed.

The PRO has taken an SRV handswitch to "OPEN" and is monitoring tailpipe temperature to confirm the SRV is open.

Select the temperature which best approximates the expected tailpipe temperature for an open SRV with the above conditions. (Assume the suppression pool is at atmospheric pressure)

A. 548°F

B. 428°F

C. 300°F

D. 212°F

Answer: C

Answer Explanation	
a.	Incorrect: Plausible to the candidate who assumes that since a BWR is a saturated system, the temperature of the steam at the discharge of the SRV would be at saturation temperature for 1050 psig. See C for reason
b.	Incorrect: Plausible to the candidate who misreads the steam table or Mollier Diagram
c	Correct: When an SRV lifts, the enthalpy contained in the steam does not change when it passes through the SRV. However, due to the lower pressure in the downcomer and suppression pool (the questions assumes atmospheric), the temperature of the steam will drop from approximatley 550°F to 300°F.
d	Incorrect: Plausible to the candidate who assumes that the suppression pool is a saturated system and therefore steam discharged into the pool would immediately cool that temperature of 212°F.

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:	0.00
System ID:	1744293
User-Defined ID:	Q #14 NEW
Lesson Plan Objective:	LGSOPS0050.7.A
Topic:	Indications of Open SRV
RO Importance:	3.4
SRO Importance:	3.6
K/A Number:	218000 A1.01

Comments:

General Data			
Level	RO		
Tier	2		
Group	1		
KA # and Rating	218000 A1.01 3.4 / 3.6		
KA Statement	Ability to predict and/or monitor changes in parameters associated with operating the AUTOMATIC DEPRESSURIZATION SYSTEM controls including: ADS valve tail pipe temperatures		
Cognitive level	Higher		
Safety Function	3 - Reactor Pressure Control		
10 CFR 55	41.5		
Technical Reference with Revision No:	Steam Tables	**Rev #:**	
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	Steam Tables		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

Reference used to answer: Mollier Diagram

Lesson Plan: GF Thermodynamics

K/A Reference: 218000, A1.01

15	**ID: 2027153**	**Points: 1.00**

Unit 1 is operating at 60%
- The 1A Service Water Pump is unavailable
- The 1B and 1C Service Water Pumps are in service when the 1C trips.
- Annunciator 106-K3, GENERATOR ROTOR HI TEMP alarms and EO investigation confirms rising temps

Which of the following identifies (1) what action is required for the given conditions and (2) the reason for the action?

A. (1) Valve out the spare TECW Heat Exchanger
(2) In order to raise cooling flow to Service Water Loads

B. (1) Valve out the spare TECW Heat Exchanger
(2) In order to lower SW Pump dP

C. (1) Secure Feedwater Heater Access Area Unit Coolers
(2) In order to raise cooling flow to Service Water Loads

D. (1) Secure Feedwater Heater Access Area Unit Coolers
(2) In order to lower SW Pump dP

Answer: A

Answer Explanation	
A	Correct: the stem indicates a trip of one of the two running SW pumps. SE-25, Degraded Service Water Capability, is entered for trip of the running pump and high temperatures of components cooled by service water. The options presented by the procedure are to start another cooling pump, which is unavailable. The next option is to secure flow to out of service components to increase cooling to in service loads. TECW out of service Heat Exchanger is one of the components directed to be secured
B	Wrong: Plausible if the candidate identifies the reverse response on pump dP for lowering system flow
C	Wrong: Plausible if the candidate does not realize that the Feedwater Heater Access Area Unit Coolers are cooled by Drywell Chill Water not Service Water
D	Wrong: Plausible if the candidate does not realize that the Feedwater Heater Access Area Unit Coolers are cooled by Drywell Chill Water not Service Water and if the student identifies the reverse response on pump dP for lowering system flow

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:	0.00
System ID:	2027153
User-Defined ID:	Q #15
Lesson Plan Objective:	LGSOPS0010.10
Topic:	Partial Loss of Service Water
RO Importance:	2.9
SRO Importance:	3.0
K/A Number:	400000 AA2.03

Comments:

General Data			
Level	RO		
Tier	1		
Group	1		
KA # and Rating	400000 AA2.03 2.9 / 3.0		
KA Statement	**Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:** High/low CCW temperature		
Cognitive level	Higher		
10 CFR 55	41.5		
Technical Reference with Revision No:	LGSOPS0010 SE-25	**Rev #:**	004 6
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)			
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	none		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

16	ID: 1797416	Points: 1.00

Unit 1 was operating at 100% when a transient occurred resulting in the following:

Reactor Power is 25%
Reactor Level is -90"
T-221, MSIV Isolation Bypass Procedure, is complete

Five minutes later, a loss of 1B-Y160 occurs.

T-270 is directed by TRIPs and Rx level is stabilized at -175"

Given:
T-251, Establish A HPIC Injection Flow Path Via Feedwater Only
T-245, RPV Injection From RHR S/D Cooling

WHICH ONE of the following identifies 1) the status of the MSIVs and 2) Secondary T-200 Procedure that will assist in RPV level control?

	MSIV Status	Secondary T-200 Procedure for RPV Level Control
A.	Open	T-251
B.	Open	T-245
C.	Closed	T-251
D.	Closed	T-245

Answer: C

Answer Explanation

With a loss of 1B-Y160, T-221 will be ineffective in preventing the MSIVs from closing on low RPV Level. This is identified in E-1BY160 step 1.23:

1.23

Precaution: Loss of 1BY160 prevents T221 and T245 from being able to be performed.

When RPV Level lowers below -129" all MSIVs will close. From LG-19 of T-117; HPCI thru Feedwater only using T-251 and RHR S/D Cooling - Through HTX ASAP (T-245)

Again from the E-1BY160 step above T-245 will not work.

Maintain RPV level between -186" AND level to which it was lowered **EXCEEDING** pump NPSH AND vortex limits if necessary

- Condensate/Feedwater – Defeat high RPV level (T-239)
- CRD
- HPCI -Thru Feedwater ONLY (T-251)
  - CST preferred
  - Defeat high RPV level (T-239)
  - Defeat high area temp (T-249)
  - Defeat high Supp Pool swapover (T-246)
  - Flow above NPSH OR vortex limits may result in pump damage
  - Min 2200 RPM
  - Max 170°F suct temp
  - Operation with Supp Pool level <18 ft increases Supp Pool press
- RCIC - CST preferred
  - Defeat high RPV level (T-239)
  - Defeat high exhaust pressure (T-238)
  - Defeat high area temp (T-249)
  - Defeat low RPV press (T-247)
  - Flow above NPSH OR vortex limits may result in pump damage
  - Min 2200 RPM
  - Max 170°F suct temp (long term) 250°F suct temp (≤ 24 hours)
  - RCIC trips on 50 psig exhaust press
- RHR S/D Cooling - Thru htx ASAP (T-245)
  - Flow above NPSH OR vortex limits may result in pump damage
- SLC Boron Tank (S48.1.B, App. 1)

LQ-19

A Wrong - plausible to the candidate that fails to recall the fact that T-221 will not work with the loss of 1B-Y160.

B Wrong - plausible to the candidate that fails to recall the fact that T-221 and T-245 will not work with the loss of 1B-Y160.

C Correct for the above reasons

D Wrong - plausible to the candidate that fails to recall the fact that T-245 will not work with the loss of 1B-Y160.

Question 16 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1797416
User-Defined ID:	Q #16
Lesson Plan Objective:	LLOT0180.08
Topic:	Impact of loss of AC power on NSSSS
RO Importance:	3.2
SRO Importance:	
K/A Number:	223002 A2.01

Comments:

General Data			
Level	RO		
Tier	2		
Group	1		
KA # and Rating	223002 A2.01 RO 3.2		
KA Statement	223002 PCIS/Nuclear Steam Supply Shutoff A2.01 - Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. electrical distribution failures		
Cognitive level	High		
Safety Function	5 - Containment Integrity		
10 CFR 55	41.5		
Technical Reference with Revision No:	E-1BY160 E-0032 Sheet 1	**Rev #:**	27 65
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	N/A		
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

17	ID: 2029824	Points: 1.00

Unit 1 is in a startup with the following conditions:

- Reactor Pressure is 150 psig
- Reactor Power is 5%
- 1B RHR is blocked out of service with a Tag Out

Consider the following sequence of events:

The 10 Bus experiences a loss of power

One minute later - the following two events occur simultaneously:

- Drywell Pressure rapidly rises to 2 psig
- The 20 Bus experiences a loss of power

WHICH ONE of the following identifies the order in which the LPCI Loops will begin injecting into the RPV?

A. 1A and 1C simultaneously followed by 1D

B. 1A and 1C and 1D simultaneously

C. 1C followed by 1A followed by 1D

D. 1C and 1D simultaneously followed by 1A

Answer: A

Answer Explanation

LOCA LOAD Sequence

t = 0 LOCA signal - Diesel starts
t = 0 Load shed
t = 0 C & D RHR pumps start
t = 3 L.C. Transformer Breaker closes
t = 5 A & B RHR pumps start
t = 10 A & C Core Spray pumps start
t = 15 B & D Core Spray pumps start

LOCA w/LOOP

t = 0 D/G Breaker closes
t = 0 All RHR pumps start
t = 3 Load Center Breakers close
t = 7 All Core Spray pumps start

From the stem, reactor pressure is below 455 psig (1/2 of the LOCA signal).

A loss of the 10 Bus will result in a loss of the 101 Bus. D11 and D13 Buses are aligned to the 101 Bus. Upon a loss of the 101 Bus, after 0.5 seconds the associated EDG will receive a start signal and the 201 breaker will close re-powering the D11 and D13 Buses after 1 second.

At this point all four divisional safegard buses are aligned to the 201 Bus with D11 EDG and D13 EDG running.

The next simultaneous event is a loss of the 20 bus which will result in a loss of the 201 Bus. As a result the D11 and D13 Buses will re-power from their associated EDG following a 0.5 second time delay. At the same time as the 20 Bus loss drywell pressure rapidly rises to 2 psig (above the 2nd half of the required LOCA signal of 1.68 psig). With this situation all LPCI pumps will start and inject as soon as power to the bus is available. For the D14 Bus the associated EDG was not already running resulting in a 10 second delay (the required time for the EDG to start and reach rated voltage and frequency) to power the D14 Bus.

The expected sequence of event is:
The 1A LPCI Loop starts at T=0.5 second
The 1C LPCI Loop starts at T=0.5 second
and the 1D LPCI Loop starting at T=~10 seconds.

A Correct for the above reasons
B Wrong - plausible to the candidate the believes that all the requirements for a LOCA/LOOP signal are met and all three will start at time 1 seconds. This would requirethe D14 EDG to already be running
C Wrong - plausible to the candidate that recognizes that the 1A and 1C RHR Pumps will have power sooner than the 1D pump but erroneously applies the LOCA loading sequence where the 1C pump starts before the 1A RHR Pump
D Wrong - plausible to the candidate that fails take into consideration the loss of the offsite power sources and just applies the LOCA loading sequence.

Question 17 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	2029824
User-Defined ID:	Q #17
Lesson Plan Objective:	LGSOPS0051.06
Topic:	LPCI - Loop Selection
RO Importance:	4.2
SRO Importance:	
K/A Number:	203000 A3.07

Comments:	General Data			
	Level	RO		
	Tier	2		
	Group	1		
	KA # and Rating	203000 A3.07 RO Importance 4.2		
	KA Statement	203000 RHR/LPCI: Injection Mode A3.07 - Ability to monitor automatic operations of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) including: Loop selection: Plant-Specific		
	Cognitive level	High		
	Safety Function	2- Reactor Water Inventory control		
	10 CFR 55	41.7		
	Technical Reference with Revision No:	LGSOPS0092A	Rev #:	
	Justification for Non SRO CFR Link:	N/A		
	Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
	Question Source: (i.e. New, Bank, Modified)	New		
	Low KA Justification (if required):	N/A		
	Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
	ILT			
	Supplied Ref (If appropriate): (i.e. ABN-##)	None		
	LORT			
	PRA: (i.e. Yes or No or #)			
	LORT Question Section: (i.e, A-Systems or B-Procedures)			
	Comments			

18	ID: 2029736	Points: 1.00

Unit 2 plant conditions are as follows:

- An ATWS is in progress
- RRCS has been initiated
- At time 0630, SLC Tank level is 3700 gallons
- At time 0650, SLC Tank level is 2875 gallons

WHICH ONE of the following identifies how many pumps are injecting at Tech Spec rated flow and at what SLC tank level will the SLC Pumps trip?

	Number of SLC pumps injecting	SLC tank level when SLC Pumps trip
A.	two	700 Gallons
B.	two	0 Gallons
C.	one	700 Gallons
D.	one	0 Gallons

Answer: D

Answer Explanation

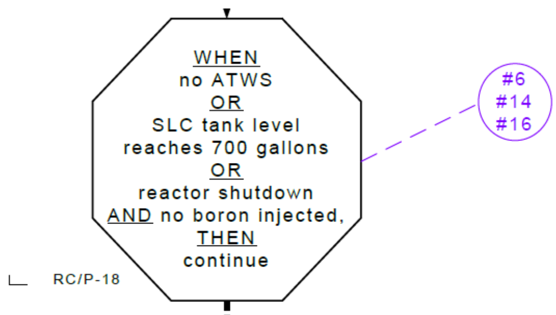
From the stem the candidate determines that SLC has been initiated automatically by the RRCS system. With the system in operation, tank level begins to decrease from the inital value of 3700 gallons to 2875 gallons 20 minutes later. The candidate should mathematically determine that the level dropped 825 gals in 20 minutes for an injection rate of ~41.2 gallons per minute. That is equal to the injection rate of a single SLC pump injecting to the reactor. 41.2 Gal/min X 20 min.= 825 gal

The injection rate is identifed in Tech Spec surveaillance 4.1.5.c as 41.2 GPM.

From the notes of T-101

#6 SLC pumps trip at 0 gallons tank level.

The below step from T-101 could lead to confusion with the candidate as to when the SLC pumps trip



A Wrong - plausible to the candidate that fails to recognise that only 1 pump is injecting based on flow rate and plausible to the candidate that incorrectly recalls that the SLC pumps automatically trip at 700 gallons (referenced in step RC/P-18)

B Wrong - plausible to the candidate that fails to recognise that only 1 pump is injecting based on flow rate.

C Wrong - plausible to the candidate that incorrectly recalls that the SLC pumps automatically trip at 700 gallons (referenced in step RC/P-18)

D Correct for the above reasons

Question 18 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	2029736
User-Defined ID:	Q #18
Lesson Plan Objective:	LGSOPS0048.12
Topic:	SLC- monitor automatic operations - Tank Level
RO Importance:	3.9
SRO Importance:	
K/A Number:	211000 A3.02

Comments:

General Data			
Level	RO		
Tier	2		
Group	1		
KA # and Rating	211000 A3.02 RO Importance 3.9		
KA Statement	211000 SLC A3.02 - Ability to monitor automatic operations of the STANDBY LIQUID CONTROL SYSTEM including: Tank level: Plant-Specific		
Cognitive level	High		
Safety Function	1 - Reactivity Conrol		
10 CFR 55	41.7		
Technical Reference with Revision No:	Tech Spec 4.1.5 T-101	**Rev #:**	2 4
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

19 **ID: 1799006** **Points: 1.00**

Unit 1 plant conditions are as follows:

- GP-2, Normal Plant Shutdown, is in progress
- Reactor Mode Switch is in START UP
- Multiple LPRMs fail upscale simultaneously resulting in the below initial APRM indications:

APRM	READING
"1"	18%
"2"	16%
"3"	14%
"4"	14%

WHICH ONE of the following identifies the RPS Scram channels that de-energize, if any?

A. None

B. RPS channel A1 de-energizes only

C. RPS channels A1 and B1 de-energize only

D. RPS channels A1, A2, B1 and B2 de-energize

Answer: D

Answer Explanation

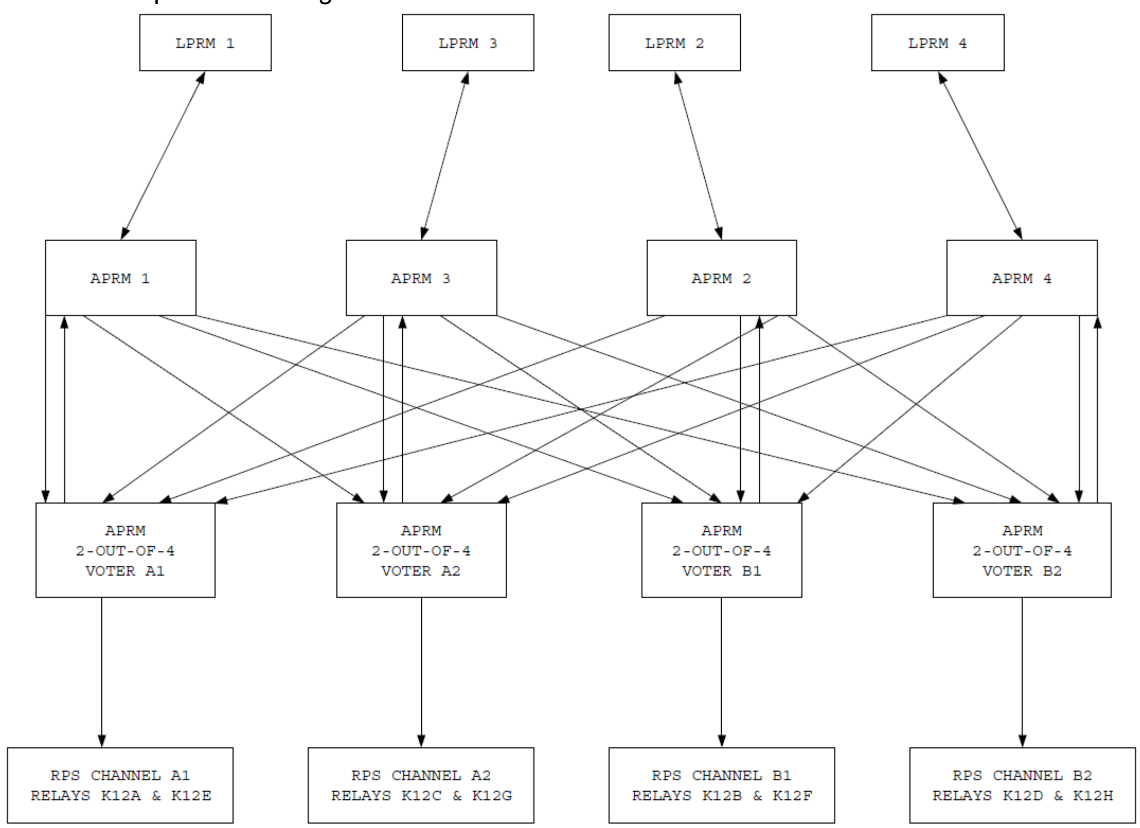
From the stem the candidate determines that APRM 1 and APRM 2 are reading above the Setdown scram value of 15 %. As a result, each APRM provides a Vote to the 4 APRM t-out-of-4 Voters. Each Voter (with two or more votes) de-energizes it's associated RPS Channel.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETP

FUNCTIONAL UNIT	TRIP SETPOINT
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale
2. Average Power Range Monitor:	
a. Neutron Flux-Upscale (Setdown)	≤ 15.0% of RATED THERMAL

From Tech Spec Bases Page B3/4 3-9:



A Wrong - Plausible to the candidate that incorrectly recalls the Scram setpoint for the Setdown portion of the APRM and believes that none of the APRMs have exceeded the Scram setpoint

B Wrong - Plausible to the candidate that incorrectly recalls the Scram setpoint for the Setdown portion of the APRM and believes that only the 1A APRM has exceeded the Scram setpoint and incorrectly associates this with the De-energization of the RPS channel A1

C Wrong - Plausible to the candidate that correctly recalls the scram setpoint for the setdown portion of the APRM and believes but believes that only the A1 and B1 RPS channels de-energize

D Correct for the above reasons

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:	0.00
System ID:	1799006
User-Defined ID:	Q #19
Lesson Plan Objective:	LGSOPS0071.04
Topic:	RPS- Ability to Monitor Power (APRMs spike due to LPRMs)
RO Importance:	4.3
SRO Importance:	
K/A Number:	212000 A4.05

Comments:

General Data		
Level	RO	
Tier	2	
Group	1	
KA # and Rating	212000 A4.05 RO Importance 4.3	
KA Statement	212000 RPS A4.05 - Ability to manually operate and/or monitor in the control room: Reactor power	
Cognitive level	Low	
Safety Function	7 - Instrumentation	
10 CFR 55	41.7	
Technical Reference with Revision No:	T. S. Table 2.2.1-1 T. S. Bases Figure B3/4.3-2	**Rev #:**
Justification for Non SRO CFR Link:	N/A	
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New	
Question Source: (i.e. New, Bank, Modified)	New	
Low KA Justification (if required):	N/A	
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	N/A	
ILT		
Supplied Ref (If appropriate): (i.e. ABN-##)	None	
LORT		
PRA: (i.e. Yes or No or #)		
LORT Question Section: (i.e, A-Systems or B-Procedures)		
Comments		

Question 19 Table-Item Links

NRC EXAM - 10 CFR 55 Operators' Licenses

CFR: 41.2 General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

CFR: 41.3 Mechanical components and design features of the reactor primary system.

CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

CFR: 41.6 Design, components, and functions of reactivity control mechanisms and instrumentation.

CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR 55.41 RO WRITTEN EXAMINATION

10 CFR 55.43 SRO WRITTEN EXAMINATION

CFR: 45.4 Identify the instrumentation systems and the significance of facility instrument readings.

CFR: 45.5 Observe and safely control the operating behavior characteristics of the facility.

CFR: 45.6 Perform control manipulations required to obtain desired operating results during normal, abnormal, and emergency situations.

CFR: 45.7 Safely operate the facility's heat removal systems, including primary coolant, emergency coolant, and decay heat removal systems, and identify the relations of the proper operation of these systems to the operation of the facility.

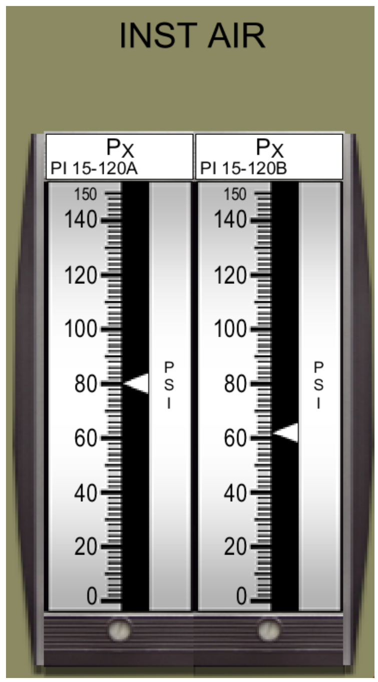
CFR: 45.8 Safely operate the facility's auxiliary and emergency systems, including operation of those controls associated with plant equipment that could affect reactivity or the release of radioactive materials to the environment.

20	ID: 1799032	Points: 1.00

Unit 1 plant conditions are as follows:

- Backup Service Air Compressor is in AUTO and aligned to Unit 1
- Service Air/Instrument Air cross-tie is aligned to the '1A' Instrument Air header
- Service Air Compressor is in AUTO

'1A' and '1B' Instrument Air header pressures drop and are now as indicated below:



WHICH ONE of the following identifies the compressor(s) supplying the U1 Instrument Air System and the compressor(s) supplying the U1 Service Air System?

	Compressor(s) supplying the U1 Instrument Air System	Compressor(s) supplying the U1 Service Air System
A.	Instrument Air Compressors -ONLY	Service Air Compressor - ONLY
B.	Instrument Air Compressors -ONLY	Service Air Compressor and Backup Service air Compressor
C.	Instrument Air Compressors and Service Air Compressor	Backup Service Air Compressor - ONLY
D.	Instrument Air Compressors and Service Air Compressor and Backup Service Air Compressor	Service Air Compressor and Backup Service Air Compressor

Answer: D

Answer Explanation
As pressure in the instrument air headers lowers the following occurs: • the instrument air compressors continue to supply the instrument air headers • When the IA header pressure drops below the SA header pressure, the SA compressor services both the IA header (for which it is aligned to) and the service air header • When SA header pressure lowers to 90 psi, the Backup Service Air Compressor starts and supports the SA header (and in this case, the 1A Instrument Air header also) • Not until pressure in both IA headers lowers below 70 psi will PV-015-*67 close. PV-015-*67 closes to isolate service air header from the service air compressor. This allows the Service Air compressor to be dedicated to supply the more vital Instrument Air header. A Wrong - Plausible if the candidate incorrectly believes normal plant air system alignment remains as the IA header pressure drops B Wrong - plausible if the candidate incorrectly recalls the pressure set point where PV-015-*67 closes and believes that PV-015-*67 closes to isolate service air header from the IA header (the opposite of the actual outcome of the closure of PV-015-*67) C Wrong - plausible if the candidate incorrectly recalls the pressure set point where PV-015-*67 closes and believes that it is closed D Correct for the above reasons

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:	0.00
System ID:	1799032
User-Defined ID:	Q #20
Lesson Plan Objective:	LGSOPS0015.02
Topic:	Monitoring of Instrument Air Pressure Gauges from MCR
RO Importance:	2.6
SRO Importance:	2.7
K/A Number:	300000 A4.01

Comments:

General Data			
Level	RO		
Tier	2		
Group	1		
KA # and Rating	300000 A4.01 2.6/2.7		
KA Statement	300000 Instrument Air System (IAS) Ability to manually operate and / or monitor in the control room: Pressure gauges		
Cognitive level	High		
Safety Function	8 - Plant Service Systems		
10 CFR 55	41.7		
Technical Reference with Revision No:	LGSOPS0015 M-0015 sh1,4	**Rev #:**	4
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Modified 2016 ILT NRC Exam		
Question Source: (i.e. New, Bank, Modified)	Modified - from bank 1153967		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	Changed 1A I/A pressure value from 60 psi to 80 psi.		
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

Question 20 Table-Item Links

NRC EXAM - 10 CFR 55 Operators' Licenses

CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

LIMERICK LO Question Category

ILT

NRC

RO

HI COG

21	ID: 1799034	Points: 1.00

Unit 1 is operating at 100% power when a loss of 1B RPS UPS Power occurs. The CRS directs the prompt performance of the below procedure steps:

2.4 **PERFORM** the following to restore Instrument Gas on 10C601, ISOLATION:

2.4.1 **PLACE** HS-59-129B, "Instrument Gas Supply" (DRYWELL B), in "CLOSE." [ ]

2.4.2 **PLACE** HS-59-102, "Instrument Gas PCIV Containment (Outboard) "in "CLOSE." [ ]

2.4.3 **PLACE** HSS-57-191B, "Containment Isolation Bypass" (B), in "BYPASS." [ ]

2.4.4 **PLACE** HS-59-129B, "Instrument Gas Supply" (DRYWELL B), in "OPEN." [ ]

2.4.5 **PLACE** HS-59-102, "Instrument Gas PCIV Containment (Outboard) "in "OPEN." [ ]

WHICH ONE of the following identifies an additional Confirming Indication of the Loss of the 1B RPS UPS Power, and the reason the above steps are to be performed promptly?

	Additional Confirming Indication	Reason for prompt performance of the above steps
A.	Loss of power to both the APRM 1/3 and APRM 2/4 Operator Display Assemblies	Restore Cooling to Recirc Pump Seal and Motor Oil Coolers
B.	Loss of power to both the APRM 1/3 and APRM 2/4 Operator Display Assemblies	Prevent MSIVs drifting closed
C.	Drywell Equipment and Floor Drain Inboard Isolation Valves close	Restore Cooling to Recirc Pump Seal and Motor Oil Coolers
D.	Drywell Equipment and Floor Drain Inboard Isolation Valves close	Prevent MSIVs drifting closed

Answer: B

Answer Explanation

From the stem the candidate concludes that procedure E-1BY160 is entered and executed. Confirming indication 1.22 reads as follows:

1.22 Loss of power to the following:

- NI-M1-1R605A, "APRM 1/3 Operator Display Assembly"
- NI-M1-1R605B, "APRM 2/4 Operator Display Assembly"
- NI-M1-1R606A, "RBM A Operator Display Assembly"
- NI-M1-1R606B, "RBM B Operator Display Assembly"

This is unique to E-1BY160 (it is not a confirming indication to E-1AY160)

The steps listed in the stem are from E-1BY160 with the a Caution above the steps that reads:

CAUTION

Nuclear Steam Supply Shutoff System (NSSSS) isolation signal must be bypassed **AND** PCIG System restored promptly to prevent Main Steam Isolation Valves (MSIVs) drifting closed on loss of gas pressure to MSIV operators.

A Wrong - plausible to the candidate that believes that the Instrument Gas system is required to support long term operation of the cooling to the Recirc seals and Motor Oil Coolers. This flow path is supported by Motor Operated Valves.

B Correct for the above reasons

C Wrong - plausible to the candidate that mistakenly recalls a Confirming Indication that actually only applies to a loss of the 1A RPS UPS Power and plausible to the candidate that believes that the Instrument Gas system is required to support long term operation of the cooling to the Recirc seals and Motor Oil Coolers. This flow path is supported by Motor Operated Valves

D Wrong - plausible to the candidate that mistakenly recalls a Confirming Indication that actually only applies to a loss of the 1A RPS UPS Power

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:	0.00
System ID:	1799034
User-Defined ID:	Q #21
Lesson Plan Objective:	LGSOPS0071.9B
Topic:	RPS/UPS - Interpreting Procedure Steps
RO Importance:	4.6
SRO Importance:	
K/A Number:	262002 2.1.20

Comments:	General Data			
	Level	RO		
	Tier	2		
	Group	1		
	KA # and Rating	262002 2.1.20 RO Importance 4.6		
	KA Statement	262002 UPS (AC/DC) 2.1.20 - Conduct of Operations: Ability to interpret and execute procedure steps.		
	Cognitive level	Low		
	Safety Function	6 - Electrical		
	10 CFR 55	41.10		
	Technical Reference with Revision No:	E-1AY160 E-1BY160	**Rev #:**	28 27
	Justification for Non SRO CFR Link:	N/A		
	Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
	Question Source: (i.e. New, Bank, Modified)	New		
	Low KA Justification (if required):	N/A		
	Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	New		
	ILT			
	Supplied Ref (If appropriate): (i.e. ABN-##)	None		
	LORT			
	PRA: (i.e. Yes or No or #)			
	LORT Question Section: (i.e, A-Systems or B-Procedures)			
	Comments			

22	ID: 1799035	Points: 1.00

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

A. SGTS Start with 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: SGTS Start with 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 candidate 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. Plausible to the candidate 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). Plausible to the candidate 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.

Question 22 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1799035
User-Defined ID:	Q #22
Lesson Plan Objective:	LLOT1560.2D
Topic:	SGTS - Recognize T-103 entry condition
RO Importance:	4.6
SRO Importance:	
K/A Number:	261000 2.4.1

Comments:

General Data			
Level	RO		
Tier	2		
Group	1		
KA # and Rating	261000 2.4.1 RO Importance 4.6		
KA Statement	261000 SGTS 2.4.1 - Emergency Procedures / Plan: Knowledge of EOP entry conditions and immediate action steps.		
Cognitive level	Low		
Safety Function	9 Radioactivity Release		
10 CFR 55	41.10		
Technical Reference with Revision No:	T-103 ARC-MCR-109, E1 ARC-MCR-002, C5 ARC-MCR-002, F2 ARC-MCR-109, B4	**Rev #:**	2 4 0 2 0 0 0 4 0 2
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	2015 CERT		
Question Source: (i.e. New, Bank, Modified)	Bank 1149969		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	added reference to SGTS starting to answer "A"		
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

Question 22 Table-Item Links

NRC EXAM - 10 CFR 55 Operators' Licenses

CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR 55.41 RO WRITTEN EXAMINATION

LIMERICK LO Question Category

ILT

NRC

RO

LOW COG

23	ID: 1799057	Points: 1.00

Plant conditions are as follows:

- A loss of offsite power has occurred
- Both units have scrammed and all rods are in
- D14 Diesel Generator has failed to start
- All other Diesel Generators are running and supplying power to their respective buses

A large break LOCA results in Unit 1 RPV level lowering to -150" and RPV pressure lowering to 250 psig.

Given the following two procedures:

S52.2.A, Core Spray System Shutdown After Automatic or Manual Initiation
S52.7.B, Core Spray Injection with a Single Operable Pump

WHICH ONE of the following identifies the automatic response of the '1B' Core Spray Pump and the follow up action to be performed?

	Automatic Response of the '1B' Core Spray Pump	Follow up action to be performed
A.	Starts 7 seconds following LOCA signal	Perform S52.2.A to minimize operation on min flow
B.	Starts 7 seconds following LOCA signal	Perform S52.7.B to prevent pump runout
C.	Starts 15 seconds following LOCA signal	Perform S52.2.A to minimize operation on min flow
D.	Starts 15 seconds following LOCA signal	Perform S52.7.B to prevent pump runout

Answer: B

Answer Explanation

From the stem the candidate determines that the station has experienced a LOOP. With a LOOP and LOCA signal present, all the Core Spray Pumps will start in 7 seconds. Since D14 D/G failed to start and offsite power is not available, all Div 4 AC loads will be de-energized. This included the '1D' Core Spray Pump, therefore only one of the two core spray pumps in the 1B Core Spray loop will be in operation and S52.7.B is appropriate to implement.

LOCA LOAD Sequence

t = 0 LOCA signal - Diesel starts
t = 0 Load shed
t = 0 C & D RHR pumps start
t = 3 L.C. Transformer Breaker closes
t = 5 A & B RHR pumps start
t = 10 A & C Core Spray pumps start
t = 15 B & D Core Spray pumps start

LOCA w/LOOP

t = 0 D/G Breaker closes
t = 0 All RHR pumps start
t = 3 Load Center Breakers close
t = 7 All Core Spray pumps start

A Wrong - plausible if the candidate confuses the power source for the injection valve, HV-52-1F037 (actually powered from Div 2).
B Correct for the above reasons
C Wrong - plausible if the candidate recalls the loading sequence for a LOCA signal due to a misconception that the LOOP and the LOCA have to happen at exactly the same time for the LOCA/LOOP loading sequence to occur and plausible if the candidate confuses the power source for the injection valve, HV-52-1F037 (actually powered from Div 2).
D Wrong - plausible if the candidate recalls the loading sequence for a LOCA signal due to a misconception that the LOOP and the LOCA have to happen at exactly the same time for the LOCA/LOOP loading sequence to occur.

Question 23 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1799057
User-Defined ID:	Q #23
Lesson Plan Objective:	LGSOPS0052 13.A
Topic:	Effect of loss of D14 Bus on 'B' Core Spray Pump during LOCA/LOOP and how to mitigate event
RO Importance:	3.4
SRO Importance:	
K/A Number:	209001 A2.03

Comments:	General Data			
	Level	RO		
	Tier	2		
	Group	1		
	KA # and Rating	209001 A2.03 RO Importance 3.4		
	KA Statement	209001 LPCS A2.03 - Ability to (a) predict the impacts of the following on the LOW PRESSURE CORE SPRAY SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. failures		
	Cognitive level	higher		
	Safety Function	2 - Reactor Water Inventory Control		
	10 CFR 55	41.5		
	Technical Reference with Revision No:	T-101 T-101 Bases LGSOPS0092A	Rev #:	3 7 2 4 0 2
	Justification for Non SRO CFR Link:	N/A		
	Question History: (i.e. LGS NRC-05, OYS CERT-04)	Modified Bank		
	Question Source: (i.e. New, Bank, Modified)	Modified Bank 833400		
	Low KA Justification (if required):	N/A		
	Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
	ILT			
	Supplied Ref (If appropriate): (i.e. ABN-##)	none		
	LORT			
	PRA: (i.e. Yes or No or #)			
	LORT Question Section: (i.e, A-Systems or B-Procedures)			
	Comments			

24	ID: 1799072	Points: 1.00

Unit 2 conditions are as follows:

- Recirc leak in Drywell is occurring
- Drywell pressure is 6.5 psig and rising up slow
- Reactor pressure is 880 psig down slow
- RCIC is injecting
- Reactor level is -98" and lowering ~ 2 inches/min
- ADS has not been inhibited

Beginning when RPV level drops below -129", which statement below describes the subsequent RPV pressure response?

RPV pressure will...

A. immediately begin to lower rapidly due to ADS valve actuation.

B. lower slowly until 105 seconds have elapsed, then lower rapidly due to ADS valve actuation.

C. lower slowly until 525 seconds have elapsed, then lower rapidly due to ADS valve actuation.

D. lower slowly over the next 10 minutes due to the LOCA condition ONLY.

Answer: B

Answer Explanation

ADS automatic valve operation requires the following logic requirements. RPV level at either –129" or the High Drywell timer initiated with a confirmatory low RPV level of 12.5". Along with the RPV low level a sequence of ECCS pumps must be operating, either the A or C RHR pumps or the A & C Core Spray pumps for Div 1, or the B or D RHR pumps or B & D Core Spray pump operation. And finally a 105 second timer initiated which following the timer ADS will initiate.

From ARC-MCR-110 A4

AUTOMATIC ACTIONS:

1. ADS will initiate in 105 seconds IF this alarm annunciates concurrently with 110 A-3 "DIV 1 ADS RELAYS ENERGIZED".

OPERATOR ACTIONS:

1. Inhibit ADS DIV 1 to prevent ADS initiation.
2. Refer to tech spec 3.3.3.

NOTE: 1. Manually securing RHR & Core Spray pumps will also prevent auto initiation.
2. IF inhibit switch is not placed in INHIBIT, THEN reset button B21A-S3A will re-start the 105 second timer provided initiation signal is still present.

CAUSES:

1. High drywell press 1.68# AND low reactor level -129" with confirmatory low level 12.5" energizing the K5A 105 second timer relay.
2. -129" Rx level for 420 seconds AND confirmatory low level 12.5" energizing the K5A 105 second timer relay.

NOTE: 1. The "A" channel has a 105 second timer and 12.5" confirmatory level permissive that the "E" channel does not.
2. Channel "A" may bring in this alarm due to excess flow check valve actuation or an instrumentation issue.
3. The ECCS pump running permissive is ("A" OR "C" RHR) OR ("A" and "C" Core Spray).
4. After the 105 second timer if 110 A-3 "DIV 1 ADS RELAYS ENERGIZED" has not annunciated by the "E" channel it will from the "A" channel (if the correct pumps are running).

A Wrong - plausible if candidate does not recall specific ADS initiation criteria
B Correct for the above reasons
C Wrong - plausible if candidate does not recall specific ADS initiation criteria
D Wrong - plausible if candidate does not recall specific ADS initiation criteria

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:	0.00
System ID:	1799072
User-Defined ID:	Q #24
Lesson Plan Objective:	LGSOPS0050.07B
Topic:	ADS operation during a LOCA - Rx Pressure Response
RO Importance:	4.2
SRO Importance:	
K/A Number:	218000.A3.08

Comments:

General Data			
Level	RO		
Tier	2		
Group	1		
KA # and Rating	218000 A3.08 importance 4.2		
KA Statement	218000 ADS A3.08 - Ability to monitor automatic operations of the AUTOMATIC DEPRESSURIZATION SYSTEM including: Reactor pressure		
Cognitive level	High		
Safety Function	3 - Reactor Pressure Control		
10 CFR 55	41.7		
Technical Reference with Revision No:	ARC-MCR-110 A4	**Rev #:**	01
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Bank PBAPS April 2013 Question #17		
Question Source: (i.e. New, Bank, Modified)	Used LGS bank # 561192 as source		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

Question 24 Table-Item Links

NRC EXAM - 10 CFR 55 Operators' Licenses

CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR 55.41 RO WRITTEN EXAMINATION

LIMERICK LO Question Category

ILT

NRC

RO

HI COG

25	ID: 1799089	Points: 1.00

Unit 1 plant conditions are as follows:

- OPCON 4
- 1A-Y160 power supply is aligned normally
- 1B-Y160 is being powered from its Primary Alternate power supply due to inverter maintenance

Then:

- An electrical fault results in the complete loss of the TSC MCC (144D-C-F)
- Shortly thereafter, a loss of Div 1 DC occurs

WHICH ONE of the following describes the effect on 1A-Y160 and 1B-Y160?

	1A-Y160	1B-Y160
A.	Energized	De-energized
B.	Energized	Energized
C.	De-energized	Energized
D.	De-energized	De-energized

Answer: D

Answer Explanation

Justification:

A. **Wrong:** Plausible if the applicant (1) does not recall that ‘1A’ RPS UPS Inverter is normally powered by DIV 1 250 VDC, or (2) believes that ‘1A’ RPS UPS Inverter static switch will successfully AUTO transfer to ‘Primary Alternate’ Source power upon the loss of DIV 1 DC (i.e., inability to recall that ‘Primary Alternate’ Source power (TSC MCC) is the same for both 1AY160 and 1BY160).

B. **Wrong:** For 1A-Y160: Plausible if the applicant (1) does not recall that ‘1A’ RPS UPS Inverter is normally powered by DIV 1 250 VDC, or (2) believes that ‘1A’ RPS UPS Inverter static switch will successfully AUTO transfer to ‘Primary Alternate’ Source power upon the loss of DIV 1 DC (i.e., inability to recall that ‘Primary Alternate’ Source power (TSC MCC) is the same for both 1AY160 and 1BY160).
For 1B-Y160: Plausible if the applicant does not recall that the TSC MCC is the ‘Primary Alternate’ Source for 1BY160, or (2) believes that automatic transfer capabilities exist between Alternate Sources when the selected Alternate Source becomes unavailable (i.e., incorrectly believes that 1BY160 would be automatically powered by its Secondary Alternate Source following loss of the TSC MCC). Manual local operator action is necessary to select between Alternate Sources.

C. **Wrong:** Plausible if the applicant does not recall that the TSC MCC is the ‘Primary Alternate’ Source for 1BY160, or (2) believes that automatic transfer capabilities exist between Alternate Sources when the selected Alternate Source becomes unavailable (i.e., incorrectly believes that 1BY160 would be automatically powered by its Secondary Alternate Source following loss of the TSC MCC). Manual local operator action is necessary to select between Alternate Sources.

D. **Correct:** The RPS UPS Inverters are normally powered from Safeguard 250 VDC. On a loss of 250 VDC, the inverter output is lost. Normally, the static switch will automatically transfer to Alternate Source power. This Alternate Source is selectable between the normally aligned ‘Primary Alternate’ and the ‘Secondary Alternate,’ which requires manual local operator action to select. Note that the ‘Primary Alternate’ Source for both the ‘1A’ and ‘1B’ RPS Buses (1AY160 & 1BY160) is the TSC UPS via the TSC MCC. A loss of RPS Bus 1BY160 occurs due to loss of the ‘Primary Alternate’ source (TSC MCC), which is powering the bus as indicated in the initial conditions. This results in a ‘B’ side Half Scram signal. RPS Bus 1AY160 (Alternate Source aligned to ‘Primary Alternate’), is initially unaffected because it is being powered by the ‘1A’ RPS UPS Inverter. The Loss of DIV 1 DC (DIV 1 250 VDC is the normal power supply to the ‘1A’ RPS UPS Inverter) would normally initiate an automatic transfer of 1AY160 Bus power to the selected Alternate Source if available (the ‘Primary Alternate’ in this case). However, the ‘1A’ RPS UPS Inverter static switch will not AUTO transfer to ‘Primary Alternate’ because it is unavailable. The result is a loss of all power to RPS Bus 1AY160. With both 1AY160 and 1BY160 de-energized, a Full Scram occurs.

Question 25 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1799089
User-Defined ID:	Q #25
Lesson Plan Objective:	LGSOPS0071 2.F
Topic:	UPS - Transfer from preferred power to alternate power supplies
RO Importance:	3.1
SRO Importance:	3.4
K/A Number:	262002 K4.01

Comments:	

General Data			
Level	RO		
Tier	2		
Group	1		
KA # and Rating	262002 K4.01 RO Importance 3.1		
KA Statement	262002 UPS (AC/DC) K4.01 - Knowledge of UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) design feature(s) and/or interlocks which provide for the following: Transfer from preferred power to alternate power supplies		
Cognitive level	High		
Safety Function	5 - Electrical		
10 CFR 55	41.7		
Technical Reference with Revision No:	ARC-MCR-120 A5 ARC-MCR-122 A5	**Rev #:**	0 0 1 0 0 1
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	LGS NRC 2012		
Question Source: (i.e. New, Bank, Modified)	Bank 1097855		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			

ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	None

LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

26	ID: 1799092	Points: 1.00

Unit 2 is at 100% power when the following power failures are experienced:

At 1200: Trip of: 2AY160 CKT 3, "2A APRM UPS INVERTER 2AD185"
At 1210: Trip of: 2PP01 CKT 21, "2A APRM UPS INVERTER 2AD185 29-A10821"

At what time did Unit 2 experience a half scram and what annunciator/indication serves as an additional confirmation identified by procedure E-2AY185?

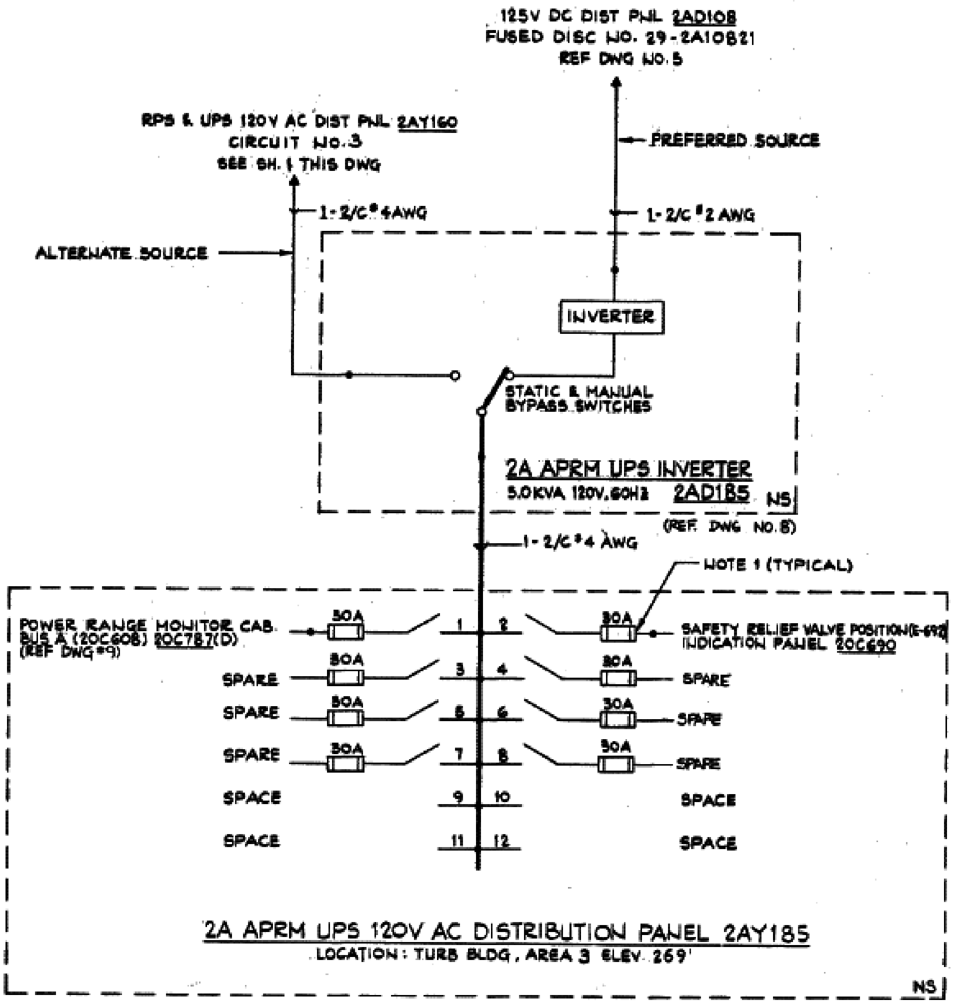
	Time half scram was received	Additional confirmation
A.	1200	ARC-MCR-210, B2, "SAFETY RELIEF VALVE OPEN"
B.	1200	Loss of Full Core Display RED (FULL OUT) and GREEN (FULL IN) indication lights
C.	1210	ARC-MCR-210, B2, "SAFETY RELIEF VALVE OPEN"
D.	1210	Loss of Full Core Display RED (FULL OUT) and GREEN (FULL IN) indication lights

Answer: C

Answer Explanation

From the stem the candidate determines that the Alternate source (From the A side RPS/UPS) to the 2A APRM UPS Inverter (2AD185) has tripped.

From E-0032 Sheet 2



This will cause alarm ARC-MCR-221 F3, 2A APRM UPS INVERTER TROUBLE, however all loads from distribution panel 2A-Y185 will remain energized.
When the preferred source to the inverter trips (2PP01 CKT 21), normally the static switch would respond by automatically transferring to the Alternate, however in this situation the Alternate Source is de-energized causing a loss of all loads on distribution panel 2A-Y185. When this occurs, APRM 2 out of 4 voters 2 and 4 will loose power causing an A side RPS half scram. A loss of distribution panel 2A-Y185 is addressed by procedure E-2AY185. An additional confirming indication in this procedure is receipt of ARC-MCR-221, F5, "2A APRM UPS INVERTER TROUBLE"

From E-2AY185:

E-2AY185 LOSS OF 2A APRM UPS POWER

1.0 CONFIRMING INDICATIONS

1.1 Any of the following Main Control Room (MCR) alarms:

- ARC-MCR-221, F5, "2A APRM UPS INVERTER TROUBLE"
- ARC-MCR 210, B2, "SAFETY RELIEF VALVE OPEN"

  AND

  ARC-MCR-207, B2, "SRV ACOUSTIC MONITOR POWER LOSS OR CABLE

Question 26 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1799092
User-Defined ID:	Q #26
Lesson Plan Objective:	LGSOPS0074A 18
Topic:	APRM/LPRM knowledge of effect from a malfunction of RPS
RO Importance:	3.7
SRO Importance:	
K/A Number:	215005 K6.01

Comments:

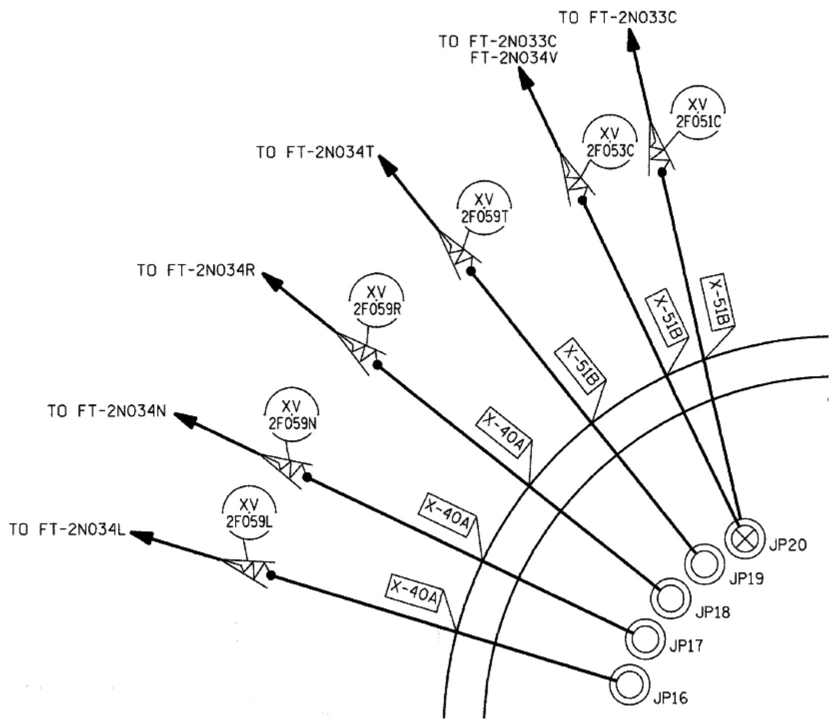
General Data			
Level	RO		
Tier	2		
Group	1		
KA # and Rating	215005 K6.01 RO Importance 3.7		
KA Statement	215005 APRM / LPRM K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM : RPS		
Cognitive level	High		
Safety Function	7 - Instrumentation		
10 CFR 55	41.7		
Technical Reference with Revision No:	E-0032 Sheet 2 ARC-MCR-221 F5 E-2AY185 E-2BY185	**Rev #:**	4 0 0 0 0 1 0 1
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

27 ID: 1799112 Points: 1.00

Unit 2 is at 100% when the following occurs:

A leak develops from the Low Pressure sensing line of Jet pump 18 and the Excess Flow Check Valve for that line functions to isolate the leak and maintain RPV integrity.

WHICH ONE of the choices below identifies the impact on the indicated total core flow and the Recirc Loop that will experience the indicated flow change?



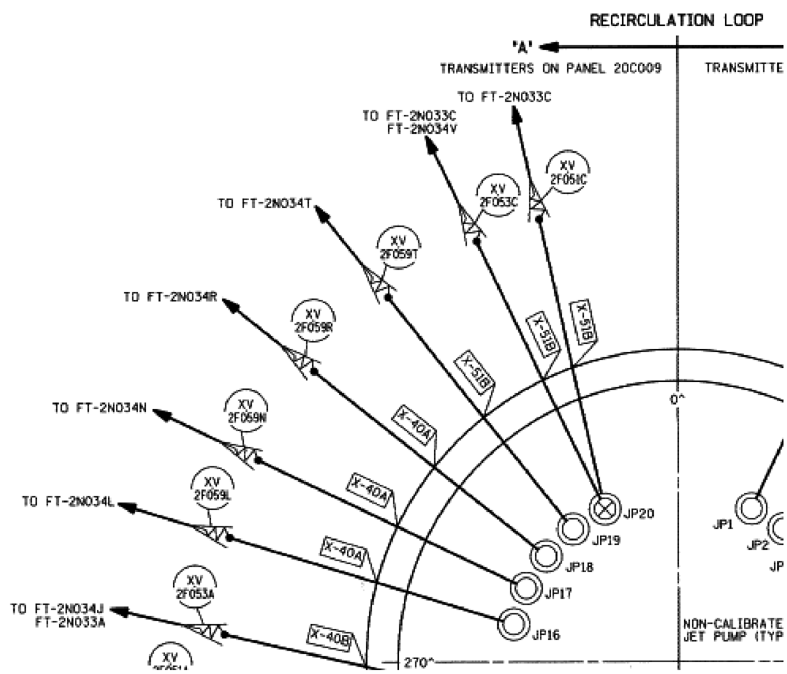
	Impact on indicated Total Core Flow	Recirc Loop experiencing indicated flow change
A.	Rises	A
B.	Rises	B
C.	Lower	A
D.	Lower	B

Answer: A

Answer Explanation

From the stem the candidate identifies that with a break in the low pressure potion of the Jet pump instrumentation the dP instrumentation for Jet Pump 18 will see a very high dP. This high dP will be converted into a very high jet pump flow signal. As a result of this high signal, indicated total core flow rises. Jet pumps 11 - 20 are associated with Recirc Loop A.

From M-0042 Sheet 4.



A Correct for the above reasons

B Wrong - plausible to the candidate the incorrectly recalls Jet Pump number assignments a belives Jet Pumps 1-10 are associated with the A Loop

C Wrong - plausible to the candidate the incorrectly recalls the response of the flow instrumentation to the Excess Flow Check Valve closure and concludes that indiated flow would go down

D Wrong - plausible to the candidate the incorrectly recalls the response of the flow instrumentation to the Excess Flow Check Valve closure and concludes that indiated flow would go down and plausible to the candidate the incorrectly recalls Jet Pump number assignments a belives Jet Pumps 1-10 are associated with the A Loop

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:	0.00
System ID:	1799112
User-Defined ID:	Q #27
Lesson Plan Objective:	LGSOPS0042
Topic:	Nuclear Instrumentation Connection to RPV - cause-effect
RO Importance:	3.6
SRO Importance:	
K/A Number:	216000 K1.22

Comments:	
	General Data

General Data			
Level	RO		
Tier	2		
Group	2		
KA # and Rating	216000 K1.22 RO Importance 3.6		
KA Statement	216000 Nuclear Boiler Inst. K1.22 - Knowledge of the physical connections and/or cause- effect relationships between NUCLEAR BOILER INSTRUMENTATION and the following: Reactor vessel		
Cognitive level	High		
Safety Function	7 - Instrumentation		
10 CFR 55	41.2 to 41.9		
Technical Reference with Revision No:	M-0042 Sheet 3 M-0042 Sheet 4	**Rev #:**	21 12
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

28	ID: 1799137	Points: 1.00

Unit 1 is at 60% power when the following occurs:

An electrical fault has occurred resulting in the following alarm associated with the 1A Rod Block Monitor (RBM).

- ARC-MCR-108, D4, “RBM DOWNSCALE/TROUBLE”

Given the following procedures:

- E-1AY160, Loss of 1A RPS UPS Power
- E-1AY185, Loss of 1A APRM UPS Power

WHICH ONE of the following identifies the Event Procedure containing this annunciator as an expected alarm and whether the 1A RBM requires bypass to allow control rod withdraw?

	Event Procedure	Bypassing the 1A RBM is required
A.	E-1AY160	No
B.	E-1AY160	Yes
C.	E-1AY185	No
D.	E-1AY185	Yes

Answer: C

Answer Explanation

From the stem the candidate determines there is a loss of 1A-Y185. 1A-Y185 is one of the APRM power supplies. 1A-Y185 and its counterpart 1B-Y185 provided power to multiple Low Voltage Power Supplies (LVPS) with each LVPS receiving power from both 1A-Y185 and 1B-Y185. The LVPS provide power to the APRM/LPRM and RBM Chassis.. Upon a loss of 1A-Y185 All APRM/LPRM and RBM Chassis continue to function.

From ARC-MCR-108- D4, a loss of 1AY185, ckt 1 or 1BY185 ckt 1 will cause the RBM DOWNSCALE/TROUBLE alarm

From E-1AY185:

NOTE

Expected MCR alarms for this event are as follows: [ ]

- ARC-MCR-107, B2, " SRV ACOUSTIC MONITOR POWER LOSS OR CABLE FAULT"
- ARC-MCR-108, A1, "NEUTRON MONITORING SYSTEM TRIP"
- ARC-MCR-108, B1, "AUTO SCRAM CHANNEL A1"
- ARC-MCR-108, B2, "AUTO SCRAM CHANNEL A2"
- ARC-MCR-108, A5, “OPRM/APRM TROUBLE”
- ARC-MCR-108, D4, “RBM DOWNSCALE/TROUBLE”
- ARC-MCR-110, B2, "SAFETY RELIEF VALE OPEN"
- ARC-MCR-121, F5, "1A APRM UPS INVERTER TROUBLE"

Also from E-1AY185:

NOTE

A loss of 1AY185 has **no effect** on Rod Block Monitor (RBM) Channel "A," other than to identify a FAULT on the channel due to a loss of one of its redundant power supplies. Control rod motion is **not** inhibited by RBM Channel "A, **AND** there is, therefore, **no** need to bypass RBM Channel "A" in order to reposition control rods if required. [ ]

A Wrong - plausible to the candidate the fails to recall the power supply that would result in the isted alarm. (believes power for the RBM comes from the RPS UPS rather than the APRM UPS)

B Wrong - plausible to the candidate the fails to recall the power supply that would result in the isted alarm. (believes power for the RBM comes from the RPS UPS rather than the APRM UPS) and plausible to the candidate the believes the loss of the APRM Power supply will required the bypass of the RBM to continue rod withdraw. (Due to a false upscale condition)

C Correct for the above reasons

D Wrong - plausible to the candidate the believes the loss of the APRM Power supply will required the bypass of the RBM to continue rod withdraw. (Due to a false upscale condition)

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:	0.00
System ID:	1799137
User-Defined ID:	Q #28
Lesson Plan Objective:	LGSOPS0074B 3.D
Topic:	RBM - knowledge of APRM Power Supplies
RO Importance:	2.8
SRO Importance:	
K/A Number:	215002 K2.03

Comments:	General Data			
	Level	RO		
	Tier	2		
	Group	2		
	KA # and Rating	215002 K2.03 RO Importance 2.8		
	KA Statement	215002 RBM K2.03 - Knowledge of electrical power supplies to the following: APRM channels: BWR-3,4,5		
	Cognitive level	Low		
	Safety Function	7 - Instrumentation		
	10 CFR 55	41.7		
	Technical Reference with Revision No:	ARC-MCR-108 D4 E-1BY185	**Rev #:**	5 1
	Justification for Non SRO CFR Link:	N/A		
	Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
	Question Source: (i.e. New, Bank, Modified)	New		
	Low KA Justification (if required):	N/A		
	Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
	ILT			
	Supplied Ref (If appropriate): (i.e. ABN-##)	None		
	LORT			
	PRA: (i.e. Yes or No or #)			
	LORT Question Section: (i.e, A-Systems or B-Procedures)			
	Comments			

29	ID: 1799143	Points: 1.00

Unit 2 is at 100% power at the completion of ST-6-055-230-1, HPCI Pump, Valve and Flow Test with the following Suppression Pool Parameters:

- 2A RHR is in Suppression Pool Cooling utilizing the 0A RHRSW Pump
- Suppression Pool Temperature is 94 °F down 2 °F/hr

A logic malfunction with the 2A RHR Initiation Logic causes a 2A RHR LOCA signal to be generated:

For the above conditions, WHICH ONE of the following correctly completes the following statement about the impact on Suppression Pool Temperature?

If directed to restore the Suppression Pool Temperature cooldown, HV-051-2F024A, RHR Full Flow Test Return, can be opened (and will remain open) ___(1)___ and HV-051-2F048A, Heat Exchanger Bypass, can be closed (and will remain closed) ___(2)___.

	(1)	(2)
A.	immediately	immediately
B.	immediately	in three minutes
C.	in three minutes	immediately
D.	in three minutes	in three minutes

Answer: B

Answer Explanation

From the stem the candidate determines that the LOCA signal malfunction of the LPCI system has impacted the ability of the 2A RHR system to cool the suppression pool. The candidate then recalls that HV-051-2F024A receives a close signal (with no seal in) and HV-051-2F048A receives an open signal (with a three minutes seal in).

From S51.8.A

NOTE

1. **IF** LOCA signal occurs during Suppression Pool cooling
   OR lowering Suppression Pool level,
   THEN the following valve interlocks occur:

   - HV-51-*F024A(B), "RHR Full Flow Test Return" (SUPP POOL CLG), closes

And

NOTE

1. HV-C-51-*F048A(B), "RHR Heat Exchanger Shell Side Bypass," receives an open signal for 3 minutes after LOCA initiation.

A Wrong - plausible if the candidate fails to recall the three minutes seal in for HV-051-2F048A open signal
B Correct for the above reasons
C Wrong - plausible if the candidate incorrectly recall that the three minutes seal in applies to HV-051-2F024A instead of HV-051-2F048A
D Wrong - plausible if the candidate incorrectly recall that the three minutes seal in applies to HV-051-2F024A in addition to HV-051-2F048A

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:	0.00
System ID:	1799143
User-Defined ID:	Q #29
Lesson Plan Objective:	LGSOPS0051 9.C
Topic:	RHR Pool Cooling - suppression pool temp
RO Importance:	3.9
SRO Importance:	
K/A Number:	219000 K3.01

Comments:

General Data	
Level	RO
Tier	2
Group	2
KA # and Rating	219000 K3.01 RO Importance 3.9
KA Statement	219000 RHR/LPCI: Torus/Pool Cooling Mode K3.01 - Knowledge of the effect that a loss or malfunction of the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE will have on following: Suppression pool temperature control
Cognitive level	Low
Safety Function	5 - Containment Integrity
10 CFR 55	41.7
Technical Reference with Revision No:	S51.8.A **Rev #:** 49
Justification for Non SRO CFR Link:	N/A
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Clinton ILT 12-1 NRC Exam Q #56
Question Source: (i.e. New, Bank, Modified)	Clinton ILT 12-1 NRC Exam Q #56
Low KA Justification (if required):	N/A
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)	
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	None
LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

30	ID: 1799175	Points: 1.00

Unit 1 plant startup is in progress with the 1B condensate pump inservice.

Condensate System startup in progress

At t = 0 seconds:

- PRO places the '1A' Condensate Pump control switch in START

At t = 33 seconds:

- '1A' Condensate Pump Discharge Valve reaches 90% open and stops there (torque switch problem)

No operator action is taken.

WHICH ONE of the following describes the '1A' Condensate Pump response?

A. Immediately trips when the Discharge Valve stops.

B. Trips 2 seconds after the Discharge Valve stops.

C. Remains running with 6000 gpm through the Min Flow Valve.

D. Remains running with 7500 gpm through the Min Flow Valve.

Answer: B

Answer Explanation

Refer to S05.1.A, page 19, Step 4.22.2 NOTE, where we see that "if the Condensate Pump discharge valve is not open or (at least still) stroking open 35 seconds after the pump control switch is taken to START, the pump will trip.

'B' is correct for the above reasons. ***Trips 2 seconds after the Discharge Valve stops.*** At t = 33 seconds, there are still 2 seconds left before the T.D. relay times out (at t = 35 seconds). When it does, it sees that the discharge valve is neither open, nor stroking open; therefore, the pump trips at that time (i.e., at t = 35 seconds).

'A' is wrong. ***Immediately trips when the Discharge Valve stops.*** Plausible to the examinee who can recall neither the correct time delay (35-seconds), nor the extent to which the discharge valve must be open (90% versus fully).

'C' is wrong. ***Remains running with 6000 gpm through the Min Flow Valve.*** Plausible to the examinee who does not recall the auto-trip feature of the pump and instead believes that, so long as there is a Min Flow Valve, there is no reason to design the pump with such a trip.

'D' is wrong. ***Remains running with 7500 gpm through the Min Flow Valve.*** Plausible for the same reason as for choice 'C'; this examinee is distracted by the 6000 gpm of choice 'C' and the 7500 gpm, here (i.e., vaguely recalls that adequate min flow requires either 2500 gpm per running pump, or 7500 gpm, regardless of the number of running pumps).

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:	0.00
System ID:	1799175
User-Defined ID:	Q #30
Lesson Plan Objective:	LGSOPS0005.IL4A
Topic:	Recall Condensate Pump Auto-Trip Feature
RO Importance:	2.8
SRO Importance:	2.8
K/A Number:	256000 K4.03

Comments:	
	General Data

General Data			
Level	RO		
Tier	2		
Group	2		
KA # and Rating	256000 K4.03 2.8 / 2.8		
KA Statement	**Knowledge of REACTOR CONDENSATE SYSTEM design feature(s) and/or interlocks which provide for the following:** Condensate and/or booster pump protection		
Cognitive level	Lower		
Safety Function	2		
10 CFR 55	CFR: 41.7		
Technical Reference with Revision No:	S05.1.A, page 19, Step 4.22.2	**Rev #:**	38
Justification for Non SRO CFR Link:			
Question History: (i.e. LGS NRC-05, OYS CERT-04)			
Question Source: (i.e. New, Bank, Modified)	Bank 558666		
Low KA Justification (if required):			
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

From LGSOPS0005.IL4A and KA 256000 K4.03

31	ID: 2029590	Points: 1.00

WHICH ONE of the following correctly completes the statements below about the "*A(B) Rx Recirc Pump Speed Lower/Raise" pushbuttons at *0C602?

If the pushbutton is held for anytime greater than _____(1)_____ seconds, a "Stuck Button" alarm will be generated and the pushbutton backlight will remain illuminated.

Upon release of the pushbutton that received the "Stuck Button" alarm, if the pushbutton is subsequently depressed again the associated ASD speed _____(2)_____ change.

	(1)	(2)
A.	5	will NOT
B.	5	will
C.	10	will NOT
D.	10	will

Answer: C

Answer Explanation

From LGSOPS0043B

2. Flow Control
   a. Recirculation flow rate is normally manually controlled by controlling recirc pump speed via the ASD using the pushbuttons at MCR panel 10-C602.
      1) Raise – 1, 5 & 10 rpm
      2) Lower – 1, 5 & 30 rpm
      3) Pushbuttons backlight when pushed acknowledging that the signal has been received
      4) If a button is depressed for 10 seconds it will result in a stuck button alarm. The button will remain back lit and must be reset in order to function. All other buttons will remain functional after the initial 10 seconds.

Conservative Bias to Plant Conditions

Reactivity Management

- Acknowledge Operations own reactivity management
- Plan and brief all reactivity changes prior to execution

Objective IL6.a

Recirc ASD Minor Failure Alarm

From S43.1.F and typical for all 6 pushbutton errors

2-00	Raise Low PB Stuck	• ALARM	• The ASD Low Raise PB on *0C602 panel was held down for more than 10 seconds or is stuck. Control room operator can verify this by checking PB-043-*08A(B)-3 is backlit **AND** remains illuminated.	1. Once the cause is correc can be cleared at *0C60 depressing **AND** holding Reset PB-043-*07A(B) f seconds.

In addition to the feedback provided from the ASD machine that the signal was recevied from the pushbutton (the normal purpose of the pushbutton backlight) the secondary purpose of the backlight feedback is that ASD machine has been receiveing a signal from the pushbutton for 10 seconds or greater and the ASD is going to ignore any subsequent signals from the pushbutton until the Fault Reset Pushbutton is depressed for 6 to 10 seconds.

A Wrong: plausible to the candidate that fails to recall that a stuck button error does not occur until 10 seconds

B Wrong: plausible to the candidate that fails to recall that a stuck button error does not occur until 10 seconds and plausible to the candidate that fails to recall that you must hold the Fault Reset Pushbutton for 6 to 10 seconds before the function of the pushbutton is restored.

C Correct for the above reasons.

D Wrong: plausible to the candidate that fails to recall that you must hold the Fault Reset Pushbutton for 6 to 10 seconds before the function of the pushbutton is restored.

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:	0.00
System ID:	2029590
User-Defined ID:	Q #31
Lesson Plan Objective:	LGSOPS0043B.6.A
Topic:	ASD Pushbutton Feedback
RO Importance:	2.6
SRO Importance:	2.6
K/A Number:	202002 K5.02

Comments:	

General Data			
Level	RO		
Tier	2		
Group	2		
KA # and Rating	202002 K5.02 2.6 / 2.6		
KA Statement	Knowledge of the operational implications of the following concepts as they apply to RECIRCULATION FLOW CONTROL SYSTEM : Feedback signals		
Cognitive level	Lower		
Safety Function	1 - Reactivity Control		
10 CFR 55	41.5		
Technical Reference with Revision No:	LGSOPS0043B	Rev #:	6
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

32	ID: 1799183	Points: 1.00

Unit 1 is operating at 18% power with the Main Turbine at rated speed in preparation for synchronization to the grid when the following occurs:

- Main Seal Oil Pump (MSOP) trips on overload
- Emergency Seal Oil Pump (ESOP) will not start (automatically or manually)

WHICH ONE of the following describes the operational implication of this event, and why?

A. Main Turbine can remain operating because the Main Generator is not yet on the grid.

B. Main Turbine can remain operating because the Main Generator casing has not yet been purged.

C. Main Turbine must be tripped because bearing oil pressure is rapidly dropping.

D. Main Turbine must be tripped because operators will have to rapidly vent off hydrogen from the Main Generator.

Answer: D

Answer Explanation

A loss of both the MSOP and ESOP leaves the bearing oil header directly supplying the generator seals at a pressure of only 8 to 10 psig. At 18% power (actually, before plant startup), the main generator has already been filled with hydrogen to a casing pressure of 75 psig. With such a large delta-P between the casing pressure and seal pressure, hydrogen will rapidly leak out of the machine. As such, ON-126, step 2.3.2 directs us to trip the main turbine,

2.3 **IF** Main Generator Hydrogen Pressure **cannot** be maintained above 55 psig
THEN PERFORM the following:

2.3.1 **IF** Reactor power is greater than 25%,
THEN INITIATE plant shutdown per GP-4,
AND ENSURE the Main Turbine is tripped. [ ]

2.3.2 **IF** Reactor power is less than 25%,
THEN ENSURE the Main Turbine is tripped. [ ]

and then step 2.10 directs us to vent off generator hydrogen (to a pressure of 5 to 8 psig).

2.10 **IF** Generator was taken off line in accordance with step 2.3.1
OR 2.3.2
THEN REDUCE Generator Gas Pressure to between 5 psig
AND 8 psig as follows:

2.10.1 **ENSURE** closed 028-*033, "H2 Excess Flow Sta. Outlet Stop Vlv." [ ]

2.10.2 **OPEN** 028-*037, "Generator H_2 Distribution Manifold Outlet Valve to Vent." [ ]

2.10.3 Slowly **OPEN** 028-*043, "Generator Gas Outlet Purge Valve to Vent." [ ]

A Wrong - Plausible to the candidate who does recognize that the generator is not yet on the grid, but who thinks this has to do with the problem of hydrogen escaping from the generator.
B Wrong - Plausible to the candidate who does comprehend the problem but who believes the generator has not yet been filled with hydrogen.
C Wrong - Plausible to the candidate who beleives that the problem is not with the seal oil system but rather with the fact that since the seal oil system is taking its supply directly from the turbine bearing oil header it robs oil away from going to the turbine bearings (it does not).
D Correct for the above reasons

Question 32 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	0.00
System ID:	1799183
User-Defined ID:	Q #32
Lesson Plan Objective:	LGSOPS0028A.IL3A
Topic:	Main Generator - impact of Loss of Both MSOP and ESOP
RO Importance:	2.8
SRO Importance:	
K/A Number:	245000 K6.03

Comments:	General Data			
	Level	RO		
	Tier	2		
	Group	2		
	KA # and Rating	245000 K6.03 RO Importance 2.8		
	KA Statement	245000 Main Turbine Gen. / Aux. K6.03 - Knowledge of the effect that a loss or malfunction of the following will have on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS : Hydrogen seal oil		
	Cognitive level	High		
	Safety Function	4 -Heat Removal From Core		
	10 CFR 55	41.7		
	Technical Reference with Revision No:	ON-126	**Rev #:**	14
	Justification for Non SRO CFR Link:	N/A		
	Question History: (i.e. LGS NRC-05, OYS CERT-04)	Bank 907635		
	Question Source: (i.e. New, Bank, Modified)	Bank 907635		
	Low KA Justification (if required):	N/A		
	Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
	ILT			
	Supplied Ref (If appropriate): (i.e. ABN-##)	None		
	LORT			
	PRA: (i.e. Yes or No or #)			
	LORT Question Section: (i.e, A-Systems or B-Procedures)			
	Comments			

33	ID: 1799184	Points: 1.00

Unit 1 plant conditions are as follows:

- Leak from 1B RHR suction piping
- Suppression Pool level is 16 feet
- A coolant leak from a recirc loop has developed
- Drywell pressure has risen to 15 psig
- Drywell and Suppression Pool sprays are attempted

WHICH ONE of the following describes the approximate value of Suppression chamber pressure as compared to Drywell pressure, based on the above conditions?

A. Equal to Drywell pressure

B. 2 psi below Drywell pressure

C. 5 psi below Drywell pressure

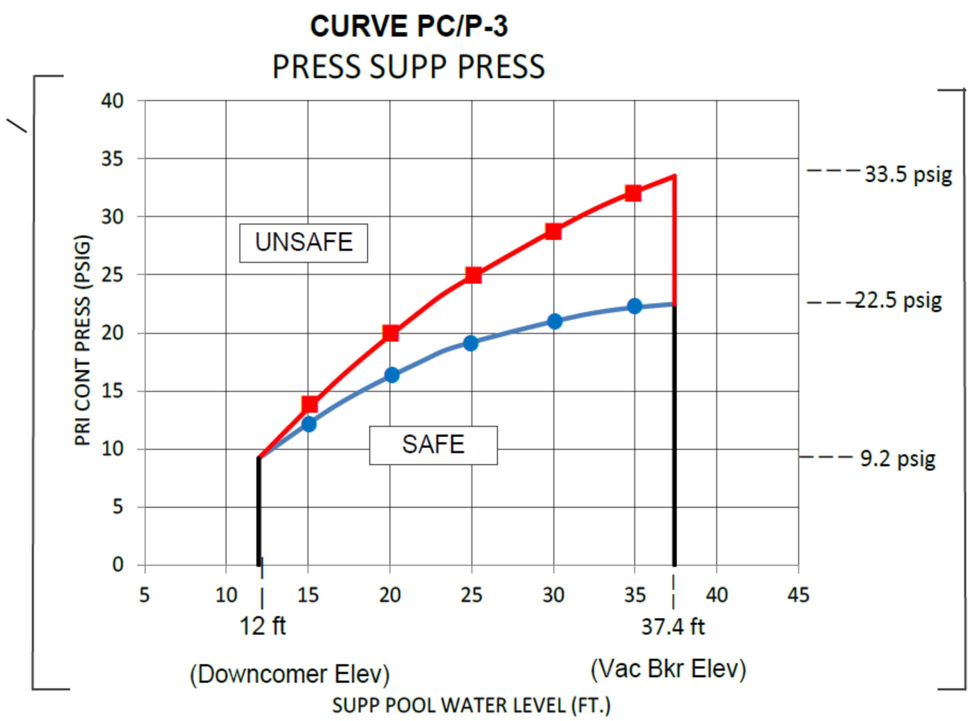
D. 8 psi below Drywell pressure

Answer: B

Answer Explanation

a. Wrong: Plausible to the candidate who confuses the minimum level for HPCI operation of 18 feet with the bottom of the drywell downcommers at 12 feet. If the downcomer were uncovered, the water seal is lost and the drywell and suppression pool would directly communicate through the downcomer. D/P = 0 Psid

b. Correct: With a lower level in the downcomer, the differential pressure required to overcome the weight of water in the Pool drops. Curve PC/P3 illustrates the effect of lowering level. At 37' in the pool, the drywell pressure would be approximately 11 pounds different. As level drops, the delta drops as well until at 12 ft, the difference is 0 psid as the pool level uncovers the downcomer.



The candidate can calculate the backpressure required by determining the water level above 12 feet in the pool (16'-12' = 4') multiplied by the pressure exerted by a column of water. ~27.7" = 1 lb.
So: 4 ft X 12" / 27.7"/lb = 1.73 # rounded to 2 psid.

c. Wrong: Plausible to the candidate who does not consider the changing pool level and only recalls the normal differential pressure experienced during a drywell leak with pool level in the normal band of approximately 5 psig. 12' delta X 12" / 27.7"/lb = 5 psid

d. Wrong: Plausible to the candidate who incorrectly calculates impact that the change in level will have and adds the change in level to the inital pool level and determines that the delta is 8ft. X 12" / 27.7"/lb = 3.4 lbs and then adds this value to the normal delta 5 + 3.4 = approx. 8 lbs.

Question 33 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1799184
User-Defined ID:	Q #33
Lesson Plan Objective:	LLOT0130.06
Topic:	Leak from RHR suction - Suppression Pool level dropped to 16 feed before the leak was isolated - A
RO Importance:	3.2
SRO Importance:	3.3
K/A Number:	223001 A1.07

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<tr><td>Tier</td><td>2</td></tr>
<tr><td>Group</td><td>2</td></tr>
<tr><td>KA # and Rating</td><td>223001 A1.07 3.2 / 3.4</td></tr>
<tr><td>KA Statement</td><td>Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES controls including: Drywell/suppression chamber differential pressure (drywell to containment building): Plant-Specific</td></tr>
<tr><td>Cognitive level</td><td>Higher</td></tr>
<tr><td>Safety Function</td><td>5</td></tr>
<tr><td>10 CFR 55</td><td>CFR: 41.5</td></tr>
<tr><td>Technical Reference with Revision No:</td><td>Rev #:</td></tr>
<tr><td>Justification for Non SRO CFR Link:</td><td></td></tr>
<tr><td>Question History: (i.e. LGS NRC-05, OYS CERT-04)</td><td></td></tr>
<tr><td>Question Source: (i.e. New, Bank, Modified)</td><td>Bank 561514</td></tr>
<tr><td>Low KA Justification (if required):</td><td></td></tr>
<tr><td>Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)</td><td></td></tr>
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<tr><th colspan="2">LORT</th></tr>
<tr><td>PRA: (i.e. Yes or No or #)</td><td></td></tr>
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Procedure Ref.: T-102
02NRC
Cognitive Level: H
LLOT0130.06
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34	ID: 2029611	Points: 1.00

Unit 1 is at 100% power with MAPRAT from the P-1 edit reading 1.07.

The Thermal Limit is being addressed in accordance with GP-14, Resolution of Thermal Limit Violations.

WHICH ONE of the following identifies (1) the impact to the nuclear fuel and (2) the Tech Spec time requirement for initiating corrective action?

	Impact to the Nuclear Fuel	Time to initiate corrective action
A.	the post-LOCA Peak Cladding Temperature is not assured to remain below design limits	15 minutes
B.	the post-LOCA Peak Cladding Temperature is not assured to remain below design limits	1 hour
C.	the requirement of less than 0.1% of fuel rods in the core being susceptible to transition boiling is not assured	15 minutes
D.	the requirement of less than 0.1% of fuel rods in the core being susceptible to transition boiling is not assured	1 hour

Answer: A

Answer Explanation

From Tech Spec Bases 3/4.2.1:

BASES

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature (PCT) following the postulated design basis Loss-of-Coolant Accident (LOCA) will not exceed the limits specified in 10 CFR 50.46 and that the fuel design analysis limits specified in NEDE-24011-P-A (Reference 2) will not be exceeded.

Mechanical Design Analysis: NRC approved methods (specified in Reference 2) are used to demonstrate that all fuel rods in a lattice operating at the bounding power history, meet the fuel design limits specified in Reference 2. No single fuel rod follows, or is capable of following, this bounding power history. This bounding power history is used as the basis for the fuel design analysis MAPLHGR limit.

LOCA Analysis: A LOCA analysis is performed in accordance with 10 CFR 50 Appendix K to demonstrate that the permissible planar power (MAPLHGR) limits comply with the ECCS limits specified in 10 CFR 50.46. The analysis is performed for the most limiting break size, break location, and single failure combination for the plant, using the evaluation model described in Reference 9.

From Tech Spec 3.2.1

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of axial location and AVERAGE PLANAR EXPOSURE shall be within limits based on applicable APLHGR limit values which have been determined by approved methodology for the respective fuel and lattice types. When hand calculations are required, the APLHGR for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) as shown in the CORE OPERATING LIMITS REPORT (COLR). During operation, the APLHGR for each fuel type shall not exceed the above values multiplied by the appropriate reduction factors for power and flow as defined in the COLR.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limiting value, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

A Correct for the above reasons

B Wrong - plausible to the candidate that fails to recall the correct response time for exceeding the APLHGR LCO

C Wrong - Plausible to the candidate the confuses the bases for MFLCPR (MCPR) with the bases for MAPRAT (APLHGR)

D Wrong - Plausible to the candidate the confuses the bases for MFLCPR (MCPR) with the bases for MAPRAT (APLHGR) plausible to the candidate that fails to recall the correct response time for exceeding the APLHGR LCO

Question 34 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	2029611
User-Defined ID:	Q #34
Lesson Plan Objective:	LGSOPS0041 6.A
Topic:	Vessel Internals - impact of Thermal Limit Violation
RO Importance:	3.7
SRO Importance:	
K/A Number:	290002 A2.05

Comments:

General Data	
Level	RO
Tier	2
Group	2
KA # and Rating	290002 A2.05
KA Statement	290002 Reactor Vessel Internals A2.05 - Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Exceeding thermal limits
Cognitive level	Low
Safety Function	5 - Containment Integrity
10 CFR 55	41.5
Technical Reference with Revision No:	Tech Spec 3.2.1 and Bases; **Rev #:**
Justification for Non SRO CFR Link:	N/A
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New
Question Source: (i.e. New, Bank, Modified)	New
Low KA Justification (if required):	N/A
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	None
LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

35	ID: 2029750	Points: 1.00

Unit 2 plant conditions:

- Reactor has scrammed
- Drywell pressure is 4.4 psig and rising slowly
- Suppression Pool Spray is in service using 2A RHR
- RPV pressure is 955 psig and steady

Five minutes later:

- RPV level dropped to -135" and has recovered to -110" and is rising slowly.

WHICH ONE of the following identifies the response of HV-51-2F027A for the above conditions and whether the MOV for HV-51-2F027A has a Thermal Overload Bypass or not?

- HV-51-2F027A, RHR Suppression Pool Spray Valve

	HV-51-2F027A	Has Thermal Overload Bypass feature
A.	Automatically closes	Yes
B.	Automatically closes	No
C.	Remains open	Yes
D.	Remains open	No

Answer: A

Answer Explanation

From the stem the candidate determines that Suppression Pool Spray is in service when a LOCA signal is received (RPV Level drops below -129"). When this occurs HV-51-2F027A receives an automatic close signal.

From OP-LG-108-101-1004, Valves. HV-51-2F027A is a valve containing the MOV Thermal Overload Bypass feature.

ATTACHMENT 2
CLASS 1E MOVs WITH THERMAL OVERLOAD BYPASS
Page 2 of 4

Valve ID	Unit	SAFETY FUNCTION	NAME	Approx S time (sec
HV-046-*26	BOTH	OPEN	RECIRC. PP. SEAL PURGE SUPPLY VENT VALVE (VENT TO RADWASTE)	19
HV-049-*F031	BOTH	BOTH	RCIC PUMP SUCTION FROM SUPP POOL PCIV (SUPP POOL)	25
HV-051-*F014A	BOTH	OPEN	*A RHR HTX. RHR S.W. INLET VLV. (*A)	92
HV-051-*F014B	BOTH	OPEN	*B RHR HTX. RHR S.W. INLET VLV. (*B)	94
HV-051-*F017A	BOTH	BOTH	*A RHR LPCI INJ PCIV (OUTBOARD A)	22
HV-051-*F017B	BOTH	BOTH	*B RHR LPCI INJ PCIV (OUTBOARD B)	22
HV-051-*F017C	BOTH	BOTH	*C RHR LPCI INJ PCIV (OUTBOARD C)	22
HV-051-*F017D	BOTH	BOTH	*D RHR LPCI INJ PCIV (OUTBOARD D)	22
HV-051-*F021A	BOTH	BOTH	*A RHR CNTMT SPRAY LINE INBOARD PCIV (INBOARD)	74
HV-051-*F021B	BOTH	BOTH	*B RHR CNTMT SPRAY LINE INBOARD PCIV (INBOARD)	74
HV-051-*F024A	BOTH	BOTH	*A RHR PP. FULL FLOW TEST RETURN VLV.	94
HV-051-*F024B	BOTH	BOTH	*B RHR PP. FULL FLOW TEST RETURN VLV.	94
HV-051-*F027A	BOTH	BOTH	*A RHR SUPP POOL SPRAY LINE PCIV	24
HV-051-*F027B	BOTH	BOTH	*B RHR SUPP POOL SPRAY LINE PCIV	24

A Correct for the above reasons
B Wrong - plausible to the candidate that fails to recall HV-51-2F027A is a valve that contains the Thermal Overload Bypass
C Wrong - plausible to the candidate that fails to recall the RHR automatic response of the RHR system to a LOCA signal (or fails to recognize the conditions for a LOCA signal are present)
D Wrong - plausible to the candidate that fails to recall the RHR automatic response of the RHR system to a LOCA signal (or fails to recognize the conditions for a LOCA signal are present) and plausible to the candidate that fails to recall HV-51-2F027A is a valve that contains the Thermal Overload Bypass

Question 35 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	2029750
User-Defined ID:	Q #35
Lesson Plan Objective:	LGSOPS0051.IL8F
Topic:	Determine RHR Valve interlocks on a LOCA
RO Importance:	3.4
SRO Importance:	
K/A Number:	226001 A3.07

Comments:	
	General Data

General Data			
Level	RO		
Tier	2		
Group	2		
KA # and Rating	226001 A3.07 3.4 / 3.3		
KA Statement	**Ability to monitor automatic operations of the RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE including:** Valve operation		
Cognitive level	Lower		
Safety Function	5 - Containment Integrity		
10 CFR 55	41.7		
Technical Reference with Revision No:	8031-M-1-E11-1040-005, sh1 8031-M-1-E11-1040-015, sh1 8031-M-1-E11-1040-016, sh1	**Rev #:**	
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank,5 Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

36	ID: 1799214	Points: 1.00

Unit 1 is at 100% power when FQRSH-061-112, Channel 3, "D/W FLR DRAIN Sump Total Flow" indicates a rise in Drywell Leakage.

IS the leakage IDENTIFIED or UNIDENTIFIED and which choice identifies a possible source of the leakage?

	IDENTIFIED or UNIDENTIFIED	POSSIBLE SOURCE
A.	UNIDENTIFIED	INPUT from the Drywell Unit Coolers
B.	UNIDENTIFIED	Recirc Pump Seal Leakage
C.	IDENTIFIED	INPUT from the Drywell Unit Coolers
D.	IDENTIFIED	Recirc Pump Seal Leakage

Answer: A

Answer Explanation

A Correct: Unidentified leakage in the drywell is considered leakage that is collected in the DW Floor Drain Sump. This includes drainage from the Drywell Floor Drains as well as Under Vessel Area and Drywell Unit Coolers Drains.

Identified Leakage is from the Recirc Pump Seals, RPV Bottom head drain, Head Seal Leakage, RPV Bellows Seal leakage and RPV Vent drains which all drain to the equipment drain tank.

Channel 3 and 4 of FQRSH-061-112 are the Flow Totalizer or Integrator channels. Channel 3 is the Floor Drain Sump Totalizer and Channel 4 is the Equipment Drain Sump Totalizer.

B Wrong: Plausible to the candidate who remembers that Floor Drain Sump is considered Unidentified Leakage but believes that Recirc Pump Seal leakage is an input to the Floor Drain Sump when it actually drains to the Equipment Drain Sump

C Wrong: Plausible to the candidate who remembers that Drywell Unit Coolers drain into the Floor Drain Sump (channel 3 of the sump integrator recorder) but believes that to be identified leakage

D Wrong: Plausible to the candidate who recalls that Recirc pump seal leakage is considered identified leakage but does not recall that it drains to the equipment drain tank, not the flooor drain sump

Question 36 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1799214
User-Defined ID:	Q #36
Lesson Plan Objective:	LGSOPS0060B.01
Topic:	DW floor drain sump integrator
RO Importance:	3.4
SRO Importance:	3.6
K/A Number:	268000 A4.01

Comments:	General Data			
	Level	RO		
	Tier	2		
	Group	2		
	KA # and Rating	268000 A4.01 3.4 / 3.6		
	KA Statement	**Ability to manually operate and/or monitor in the control room:** Sump integrators		
	Cognitive level	Lower		
	Safety Function	9 - radioactivity release		
	10 CFR 55	CFR: 41.7		
	Technical Reference with Revision No:	M-0061 Sht. 4	Rev #:	15
	Justification for Non SRO CFR Link:	N/A		
	Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
	Question Source: (i.e. New, Bank, Modified)	New		
	Low KA Justification (if required):	N/A		
	Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
	ILT			
	Supplied Ref (If appropriate): (i.e. ABN-##)	None		
	LORT			
	PRA: (i.e. Yes or No or #)			
	LORT Question Section: (i.e, A-Systems or B-Procedures)			
	Comments			

37	ID: 1799196	Points: 1.00

WHICH ONE of the following requires a Unit 1 Tech Spec entry (i.e., actual implementation of a Required Action)?

A. All IRM's on range 7 during a normal reactor startup when IRM 'D' is declared inoperable

B. All IRMs on Range 8 during a normal reactor startup when APRM '3' fails upscale

C. Mode Switch in STARTUP; reactor is at the POAH when power is lost to the RWM

D. 20% reactor power during a normal plant startup when RBM 'A' fails and is declared inoperable

Answer: C

Answer Explanation	
A	Wrong: Plausible to the candidate who does not remember that Per TS Table 3.3.1-1, only 3 IRMs per RPS Trip System are required to be OPERABLE
B	Wrong: Plausible to the candidate who does not remember that Per TS Table 3.3.1-1, only 3 (total) APRMs are required to be OPERABLE. Therefore, the OPERABILITY requirements for Trip Functions #1 and #4 of the Control Rod Block Instrumentation TS 3.3.6, Table 3.3.6-1, are satisfied,
C	Correct: Per TS 3.1.4.1, the RWM is required to be OPERABLE in OPCON 1 or 2 at $\leq$ 10% RTP. Examinee is expected to recognize that power is well below 10% RTP when the "reactor is at the POAH." This is generally 1 to 2% Rx Power The applicable action is not required to be identified by the candidate but in this case is action 3.1.4.1.a which requires a second licensed operator or technically qualified individual to verify rod movement and compliance with the rod pattern.
D	Wrong: Plausible to the candidate who does not remember that Per TS 3.1.4.3, RBM Operability is NOT required until power is at least 30%; therefore the Control Rod Block Instrumentation requirements of TS 3.3.6, Table 3.3.6-1, Trip Function #1 do not apply either

Question 37 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1799196
User-Defined ID:	Q #37
Lesson Plan Objective:	LLOT0073B.7
Topic:	Recognize Tech Spec required entry condition
RO Importance:	3.4
SRO Importance:	4.7
K/A Number:	201006 (2.2.40)

Comments:	

General Data			
Level	RO		
Tier	2		
Group	2		
KA # and Rating	201006 2.2.40 3.4 / 4.7		
KA Statement	Rod Worth Minimizer/ Ability to apply Technical Specifications for a system		
Cognitive level	Higher		
Safety Function	7		
10 CFR 55	41.10		
Technical Reference with Revision No:	LGS TS 3.1.4.1	**Rev #:**	
Justification for Non SRO CFR Link:			
Question History: (i.e. LGS NRC-05, OYS CERT-04)			
Question Source: (i.e. New, Bank, Modified)	Bank 1102524		
Low KA Justification (if required):			
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)			
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

ILT #1102524

38	ID: 2029609	Points: 1.00

A coolant leak into the Unit 1 drywell has resulted in the following:

- Drywell pressure is 20 psig up slow
- RPV level is -90" down slow
- RPV pressure is 850 psig down slow
- RHR "A" Loop Injection "INITIATION" pushbutton was armed and depressed
- "A" RHR Loop is in Drywell Spray

A fault on the 101 Bus occurs

WHICH ONE of the following correctly completes the description of the automatic response of the "A" RHR loop?

Following re-energization of the D11 Bus, the 1A RHR Pump ___(1)___ and the drywell spray valves ___(2)___.

	(1)	(2)
A.	remains shutdown	remain open
B.	remains shutdown	close
C.	restarts	remain open
D.	restarts	close

Answer: C

Answer Explanation

From the stem the candidate determines that a LOCA signal has been provided to the RHR system ("initiation" pushbutton was armed and depressed). At this time the 1A RHR Pump is being powered from the D11 bus through the 101-D11 Bus. The next event is the trip of the 101 Bus. From this the candidate determines that the D11 Bus experiences a Dead Bus condition which causes the 1A RHR Pump Breaker to trip, the D11 EDG to get a start signal (0.5 seconds after dead bus condition) and the 201-D11 Bus Breaker to close (1.0 seconds after dead bus condition). Once the D11 Bus is re-energized from the 201 bus, the 1A RHR pump will start following a time delay (5 seconds).

From LGSOPS0051

b. The RHR pumps start

   1) With off-site power available RHR pumps C&D will start immediately and RHR pumps A&B will start after a five (5) seconds delay.

   2) Without off-site power available each RHR pump starts when its associated Diesel Generator output breaker shuts.

c. Test Return Valves (F024A, B and F010A, B) receive a close signal and close if open.

d. Suppression Pool spray Valves (F027A, B) receive a close signal and close if open.

A Wrong - plausible to the candidate the fails to recognize that the RHR LOCA signal will remain and will restart the RHR Pump once power is again available. (Candidate mistakenly believes that a Core Spray System LOCA signal is required, i.e. the D11 EDG will not get a LOCA start signal)

B Wrong - plausible to the candidate the fails to recognize that the RHR LOCA signal will remain and will restart the RHR Pump once power is again available. (Candidate mistakenly believes that a Core Spray System LOCA signal is required, i.e. the D11 EDG will not get a LOCA start signal) and plausible to the candidate that confuses the response of the Drywell Spray valve with other RHR valves that auto reposition under various Initiation signals.

C Correct for the above reasons

D Wrong - plausible to the candidate that confuses the response of the Drywell Spray valve with other RHR valves that auto reposition under various Initiation signals.

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:	0.00
System ID:	2029609
User-Defined ID:	Q #38
Lesson Plan Objective:	LGSOPS0051 9.A
Topic:	RHR - Drywell Spray - Pump Starts
RO Importance:	3.5
SRO Importance:	
K/A Number:	226001 A3.07

Comments:	Bank 846285

General Data			
Level	RO		
Tier	2		
Group	2		
KA # and Rating	226001 A3.07 RO Importance 3.5		
KA Statement	226001 RHR/LPCI: CTMT Spray Mode A3.07 - Ability to monitor automatic operations of the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE including: Pump start		
Cognitive level	High		
Safety Function	5 - Containment Integrity		
10 CFR 55	41.7		
Technical Reference with Revision No:	LGSOPS0051	**Rev #:**	04
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Modified 846285		
Question Source: (i.e. New, Bank, Modified)	Modified 846285		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

39	ID: 1799231	Points: 1.00

Unit 1 is Operating at 100% power

- The 1B Reactor Recirc Pump Trips

Which P-1 parameter listed below will rise as a result?

A. MFLCPR

B. MFLPD

C. MAPRAT

D. PCMARG

Answer: A

Answer Explanation

A Correct: Maximum Fraction Of Limiting Critical Power Ratio (MFLCPR), is the Critical power ratio limit divided by the critical power ratio multiplied by a flow biasing factor. When core flow drops, CPR drops and this causes MFLCPR to rise

B Wrong: Maximum Fraction Of Limiting Power Density (MFLPD). Plausible to the candidate who confuses the changes in MFLPD with MFLCPR. Actual MFLPD value will drop as power in the core drops

C Wrong: Plausible to the candidate who confuses the changes in MAPRATwith MFLCPR. Maximum Average Planar Ratio (MAPRAT) will also drop as core power drops

D Wrong: Plausible to the candidate who confuses the changes in PCMARG with MFLCPR. PCMARG is the margin to preconditioning and is proportional to pin power. As power drops, so does PCMARG

Question 39 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1799231
User-Defined ID:	Q #39
Lesson Plan Objective:	LLOT1540.05
Topic:	Loss of Forced Circulation
RO Importance:	3.6
SRO Importance:	4.1
K/A Number:	295001 AK1.03

Comments:	General Data	
	Level	RO
	Tier	1
	Group	1
	KA # and Rating	295001 AK1.03 3.6 / 4.1
	KA Statement	**Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION :** †Thermal limits
	Cognitive level	Higher
	Safety Function	1, 4
	10 CFR 55	CFR: 41.8 to 41.10
	Technical Reference with Revision No:	OT-112, P-1 — **Rev #:** 57
	Justification for Non SRO CFR Link:	
	Question History: (i.e. LGS NRC-05, OYS CERT-04)	
	Question Source: (i.e. New, Bank, Modified)	New
	Low KA Justification (if required):	
	Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)	
	ILT	
	Supplied Ref (If appropriate): (i.e. ABN-##)	None
	LORT	
	PRA: (i.e. Yes or No or #)	
	LORT Question Section: (i.e, A-Systems or B-Procedures)	
	Comments	

40	ID: 1799234	Points: 1.00

T-111, Steam Cooling, is in progress with the following:

- Reactor water level is -207"
- No injection systems are available to the RPV

WHICH ONE of the following correctly completes the following two statements:

For the above conditions Adequate Core Cooling _____(1)_____ present.

With no injection into the RPV, adequate core cooling is defined to exist as long as peak clad temperature remains below _____(2)_____?

	(1)	(2)
A.	is	1800°F
B.	is	2200°F
C.	is NOT	1800°F
D.	is NOT	2200°F

Answer: C

Answer Explanation

LR-9 Any system, subsystem, OR alt subsystem lined up with at least one pump running

The terms **"systems", "subsystems",** and **"alternate subsystems"** have been defined previously in this Bases document, and are repeated here for clarity.

A "NO" response indicates that no RPV injection sources are available, and that steam cooling is required. Operators are directed to continue at Step LR-10, where the steam cooling section of the T-111, Level Restoration/Steam Cooling, flowchart begins.

Step LR-10 directs an exit from the RPV pressure (RC/P) control flowpath of T-101, RPV Control. The steam cooling section of T-111, Level Restoration/Steam Cooling, contains RPV pressure control steps which are in conflict with those in the RC/P flowpath of T-101. To avoid the occurrence of concurrently effective, yet conflicting, RPV pressure control guidance between T-101 and T-111, operators are directed to exit the RC/P flowpath of T-101.

Steam cooling is performed to prolong the time that adequate core cooling is assured when no RPV injection sources are available. With no injection into the RPV, adequate core cooling is defined to exist as long as peak clad temperature remains below 1800 deg, the threshold for significant metal-water reaction. The RPV level at which this occurs is designated as the Minimum Zero-Injection RPV Water Level (MZIRWL) and is -198 inches.

A wrong - plausible to the candidate that confuses the RPV level for 2/3 core coverage (-211") with the RPV level for MZIRWL.

B wrong - plausible to the candidate that confuses the RPV level for 2/3 core coverage (-211") with the RPV level for MZIRWL and plausible if the candidate mistakenly believes that the ECCS design criteria of peak cladding temperature below 2200 degrees F constitutes adequate core cooling.

C Correct for the above reasons

D Wrong - plausible if the candidate mistakenly believes that the ECCS design criteria of peak cladding temperature below 2200 degrees F constitutes adequate core cooling.

Question 40 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	0.00
System ID:	1799234
User-Defined ID:	Q #40
Lesson Plan Objective:	LGSOPS1560.03
Topic:	T-111 - Recognize if Adequate Core Cooling Exists
RO Importance:	4.6
SRO Importance:	4.7
K/A Number:	295031 EK1.01

Comments:

General Data	
Level	RO
Tier	1
Group	1
KA # and Rating	295031 EK1.01 4.6/4.7
KA Statement	**Reactor Low Water Level Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL :** EK1.01 Adequate core cooling
Cognitive level	lower
Safety Function	2 - Reactor Water Inventory Control
10 CFR 55	41.8
Technical Reference with Revision No:	T-111 — **Rev #:** 15
Justification for Non SRO CFR Link:	n/a
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Modified 1151444 (Used on LGS ILT NRC Exam January 2016)
Question Source: (i.e. New, Bank, Modified)	Modified 1151444
Low KA Justification (if required):	n/a
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)	Changed RPV Level in Stem from -178" to -207", this changes correct answer from A to C.
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	none
LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

560666

Question 40 Table-Item Links

NRC EXAM - 10 CFR 55 Operators' Licenses

CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR 55.41 RO WRITTEN EXAMINATION

LIMERICK LO Question Category

ILT

NRC

RO

LOW COG

41	ID: 1799237	Points: 1.00

Plant conditions are as follows:

- Unit 1 is at 100%.
- Unit 2 is in OPCON 5 with refueling activities in progress with secondary containment set on the refuel floor.
- All, "REFUEL FLOOR/RX ENCL CNTMT ISO INTERLOCK" switches are in "NORMAL"
- A fuel handling accident results in Refuel Floor ventilation radiation levels of 13 mR/hr

Regarding the reactor enclosure and the refuel floor, which of the following describes the Zones SBGT will maintain at a negative pressure and the reason for the initiation of SBGT?

	Zones SBGT will maintain negative	Reason for SBGT Initiation
A.	Refuel Floor ONLY	Limit iodine and particulate concentration in gases, prior to discharge
B.	Refuel Floor ONLY	Limit particulate concentration in gases ONLY, prior to discharge
C.	Unit 1 reactor enclosure and Refuel floor	Limit iodine and particulate concentration in gases, prior to discharge
D.	Unit 1 reactor enclosure and Refuel floor	Limit particulate concentration in gases ONLY, prior to discharge

Answer: A

Answer Explanation	
A	Correct Refuel HVAC isolates at 2.00 mr/h. Although refuel floor containment and Unit 1 Reactor Containment are set, only when Zones are crosstied will a refuel HVAC isolation also isolate the Reactor enclosure. The purpose of the SBGT filters per the Design basis document L-S-32 is The SGTS/RERS filters iodine and particulate concentrations in gases potentially present within the Secondary Containment prior to discharge to the environment via the North Stack.
B	Incorrect Limit particulate only is incorrect but plausible to the examinee who does not recall the purpose of the charcoal filters
C	incorrect plausible to the examinee who recognizes that the radiation levels are above the Reactor Enclosure setpoint, but either does not recall the crosstie logic or believe that hi refuel radiation will isolate the reactor enclosure as long as the zone is established
D	incorrect plausible to the examinee who recognizes that the radiation levels are above the Reactor Enclosure setpoint, but either does not recall the crosstie logic or believe that hi refuel radiation will isolate the reactor enclosure as long as the zone is established and does not recall the purpose of the charcoal filters

Question 41 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1799237
User-Defined ID:	Q #41
Lesson Plan Objective:	LLOT0076B.6
Topic:	High Off-site Release Rate - protection of the general public
RO Importance:	4.2
SRO Importance:	
K/A Number:	295038 EK1.02

Comments:	

General Data			
Level	RO		
Tier	1		
Group	1		
KA # and Rating	295038 EK1.02 RO importance 4.2		
KA Statement	295038 High Off-site Release Rate / 9 EK1.02 - Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE : Protection of the general public		
Cognitive level	Low		
Safety Function	9 - Radioactivity Release		
10 CFR 55	41.8		
Technical Reference with Revision No:	S76.1.C	**Rev #:**	15
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

42	**ID: 1799238**	**Points: 1.00**

An electrical fault has caused a loss of AC power to the DIV I Battery Charger 1BCA2.

WHICH ONE of the following describes the effect on the DIV I, 125 VDC loads?

A. Some loads are lost immediately

B. All loads are lost immediately

C. All loads are lost when the battery is fully discharged

D. Some loads are lost when the battery is fully discharged

Answer: D

Answer Explanation

Answer: Some loads are lost when the battery is fully discharged

When a battery charger becomes inoperable the batteries act as a backup to keep DC available for all DC loads. As the batteries discharge some of the DC loads will become unavailable, while others use an auctioneering circuit to an alternate source to ensure availability.

A Wrong - plausible to the candidate that fails to recall that the battery charger loss does not result in the immediate loss of DC loads

B Wrong - plausible to the candidate that fails to recall that the battery charger loss does not result in the immediate loss of DC loads and plausible to the candidate that fails to recall that there are two 125VDC batteries and two battery chargers. The 1BCA1 will continue to carry loads for approximately half of the Division 1 DC loads.

C Wrong - plausible to the candidate that fails to recall that there are two 125VDC batteries and two battery chargers. The 1BCA1 will continue to carry loads for approximately half of the Division 1 DC loads.

D Correct for the above reasons

Question 42 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1799238
User-Defined ID:	Q #42
Lesson Plan Objective:	LGSOPS0095.02A
Topic:	Partial loss of DC Power - battery chargers
RO Importance:	3.1
SRO Importance:	3.1
K/A Number:	295004 AK2.01

Comments:

General Data			
Level	RO		
Tier	1		
Group	1		
KA # and Rating	295004 AK2.01 3.1 / 3.1		
KA Statement	Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: Battery charger		
Cognitive level	Lower		
Safety Function	6 - Electrical		
10 CFR 55	41.7		
Technical Reference with Revision No:	E-0033 sh. 1	**Rev #:**	45
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Bank 833421		
Question Source: (i.e. New, Bank, Modified)	Bank 833421		
Low KA Justification (if required):			
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

43	ID: 1799263	Points: 1.00

Unit 2 is at 65% power when a fire is reported. The 2B Condensate Pump Motor is burning.

Using portions of Pre-Fire Plan F-T-266, the CRS directs de-energizing the 2B Condensate Pump.

WHICH ONE of the following identifies the breaker to be opened, and the additional transient procedure requiring entry?

	Breaker to be opened	Additional Transient Procedure
A.	21 Aux Bus-04	OT-100
B.	21 Aux Bus-04	OT-112
C.	22 Aux Bus-04	OT-100
D.	22 Aux Bus-04	OT-112

Answer: C

Answer Explanation

From the stem the candidate is informed that the 2B Condensate Pump is burning. The candidate recalls that the 2B Condensate Pump is powered from the 22 Aux Bus and determines the equipment isolation for 22 Aux Bus-04. Upon the isolation of the 2B Condensate Pump entry into OT-100 is required.

From F-T-266

PLANT SYSTEMS REQUIRING MANAGEMENT:

NOTE: The following step will be performed as directed by Shift Supervision:
De-energize OR isolate equipment based on fire scene reports.

Component	Equip. Number	Equip. Isolation (Location)
Turbine Encl Supply Fans	2AV101	214B-23 (628-T9-302)
	2BV101	224B-63 (628-T9-302)
	2CV101	214A-23 (336-A8-217)
Turbine Encl Equip Comp. Exhaust Fans	2AV106	D21-10 (429-A8-239)
	2BV106	D22-10 (431-A8-239)
Condensate Pump Room Unit Coolers	2AV112	214C-T-G-12 (354-T10-217)
	2BV112	224C-T-G-12 (354-T10-217)
	2CV112	214A-G-F-12 (465-T3-239)
	2DV112	224A-G-F-12 (465-T3-239)
	2EV112	214A-G-F-13 (465-T3-239)
	2FV112	224A-G-F-13 (465-T3-239)
Condensate Pump	2AP102	21 Aux Bus-04 (336-A8-217)
	2BP102	22 Aux Bus-04 (336-A8-217)
	2CP102	21 Aux Bus-05 (336-A8-217)
Turbine Encl Cond Pump Area Equip	2AP126	214A-G-F-04 (465-T3-239)
	2BP126	224A-G-F-04 (465-T3-239)
Drain Sump Pumps	2AP127	214A-G-F-05 (465-T3-239)
	2BP127	224A-G-F-05 (465-T3-239)

From OT-100

OT-100 REACTOR LOW LEVEL

1.0 ENTRY CONDITIONS

1.1 Condensate Pump trip
OR Reactor Feed Pump (RFP) trip

A Wrong - plausible to the candidate that incorrectly recalls the Condensate Pump power supply and selects the breaker for the 2A Condensate Pump

B Wrong - plausible to the candidate that incorrectly recalls the Condensate Pump power supply and selects the breaker for the 2A Condensate Pump and plausible to the candidate that mistakenly determines that a 42% recirc runback will occur for the given situation (this would occur if total feedwater flow was >12 Mlbm/hr) For information in the stem, the total feedwater flow is ~9.9 Mlbm/hr (full power FW flow is ~15.3 Mlbm/hr X 0.65 = 9.945)

C Correct for the above reasons

D Wrong - plausible to the candidate that mistakenly determines that a 42% recirc runback will occur for the given situation (this would occur if total feedwater flow was >12 Mlbm/hr) For information in the stem, the total feedwater flow is ~9.9 Mlbm/hr (full power FW flow is ~15.3 Mlbm/hr X 0.65 = 9.945)

Question 43 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1799263
User-Defined ID:	Q #43
Lesson Plan Objective:	LGSOPS0022 8
Topic:	Fire - impact on motors
RO Importance:	2.5
SRO Importance:	
K/A Number:	600000 AK2.03

Comments:

General Data			
Level	RO		
Tier	1		
Group	1		
KA # and Rating	600000 AK2.03 RO Importance 2.5		
KA Statement	600000 Plant Fire On-site / 8 AK2.03 - Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Motors		
Cognitive level	High		
Safety Function	8 - Plant Service Systems		
10 CFR 55	41.7		
Technical Reference with Revision No:	F-T-266 OT-100 ARC-MCR-204 F2	**Rev #:**	0 5 3 6 0 1
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

44	ID: 1799241	Points: 1.00

Unit 1 plant conditions:

- 90% power
- Both Recirc Pumps running at 75% Speed

Operators are swapping in-service Stator Water Cooling (SWC) Pumps:

- SWC trips have not been bypassed

During the swap, system flow drops to 0 gpm for a period of 12 seconds before system parameters return to normal.

WHICH ONE of the following identifies the Recirc Pump speeds?

	1A	1B
A.	75%	75%
B.	42%	75%
C.	75%	42%
D.	42%	42%

Answer: B

Answer Explanation

A Wrong: Plausible to the candidate who believes that the power level listed above is below that which would cause a runback

B Correct: From ON-114-

IF Stator current >7,469 amps
and
Loss of Stator Cooling detected by any of the following:
- Inlet pressure (U1: <45 psig, U2: < 39.4 psig)
- High Stator Water bulk outlet temperature >80 °C
- Bushing coolant flow < 84 gpm.

Then IF Total FW flow >6.7 Mlbm/hr

-Reactor recirculation Pump "A" runs back to 42% speed after 9 second time delay
AND
- *B after 18 second time delay

As a result of a mod installed during 1R15, the Recirc Pumps will receive a high-limit (42%) runback (so long as Total FW flow is >6.7 Mlbm/hr). The 9 and 18-second time delays (from the point of the SWC trip signal being generated) remain the same as before the mod.

C Wrong: Plausible to the candidate who remembers the power level required to receive a runback but believes that the B RRP will runback 1st

D Wrong: Plausible to the candidate who remembers the power level requirements but forgets the 9 and 18 second time delay or believes that both pumps will run back regardless whether the signal clears.

With the SWC trip signal existing only for 12 seconds, the 1A Recirc Pump will run back to 42% speed (i.e., >9 seconds), but the 1B Recirc Pump will remain at 75% speed (i.e., <18 seconds).

Question 44 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1799241
User-Defined ID:	Q #44
Lesson Plan Objective:	LGSOPS0033.03.A
Topic:	12 second loss of Stator Coolant
RO Importance:	3.4
SRO Importance:	3.6
K/A Number:	295018 AK2.02

Comments:

General Data			
Level	RO		
Tier	1		
Group	1		
KA # and Rating	295018 AK2.02 3.4 / 3.6		
KA Statement	**Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following:** Plant operations		
Cognitive level	Higher		
Safety Function	8 - Plant Service Systems		
10 CFR 55	CFR: 41.7		
Technical Reference with Revision No:	ON-114	**Rev #:**	46
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Bank 988524		
Question Source: (i.e. New, Bank, Modified)	Bank 988524		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

45	ID: 1799264	Points: 1.00

Unit 2 is shutdown for a forced outage following a 192 day run.

- Reactor coolant temperature is 177°F in OPCON 4
- 2A RHR Pump is running in Shutdown Cooling
- HV-51-2F009, RHR SHUTDOWN CLG SUCTION INBRD PCIV (INBOARD), Fails Closed and cannot be reopened
- ON-121, Loss of Shutdown Cooling, is entered

Which of the following methods of Decay Heat Removal is available?

A. Swap to the 2B RHR Pump and restore Shutdown Cooling

B. Swap to the 2C RHR Pump and restore Shutdown Cooling

C. Restore cooling by placing RHR in ADHR

D. Use SRVs and Suppression Pool Cooling as Alternate SDC

Answer: D

Answer Explanation

A Wrong: Plausible to the candidate who believes that Swapping to the 2B pump would use a different flowpath. The HV-51-2F009 valve is failed closed and cannot be reopened. This path is not available

B Wrong: Plausible to the candidate who believes that Swapping to the 2C pump would allow injection on a different flow path.

C Wrong:restore cooling by placing RHR in ADHR is plausible to the candidate who recalls that ADHR has a different flowpath then SDC through the F009 Valve. However, this flowpath is only available when the reactor is flooded up.

D Correct: From ON-121, step 2.1.9 "IF required to implement Alternate Shutdown Cooling due to the
failure of HV-051-*F008 OR HV-051-*F009 THEN PERFORM Attachment 6". Attachment 6 then directs the operator to perform S41.7.B

Question 45 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1799264
User-Defined ID:	Q #45
Lesson Plan Objective:	LGSOPS1550.01
Topic:	Feed and Bleed
RO Importance:	3.3
SRO Importance:	3.4
K/A Number:	295021 AK3.02

Comments:	

General Data	
Level	RO
Tier	1
Group	1
KA # and Rating	295021 AK3.02 3.3 / 3.4
KA Statement	**Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING :** Feeding and bleeding reactor vessel
Cognitive level	Higher
Safety Function	4
10 CFR 55	CFR: 41.5
Technical Reference with Revision No:	S41.7.B **Rev #:** 8
Justification for Non SRO CFR Link:	
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Perry 2015 NRC Exam RO #5
Question Source: (i.e. New, Bank, Modified)	Modified
Low KA Justification (if required):	
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	None
LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

46 **ID: 1799276** **Points: 1.00**

Unit 1 plant conditions are as follows:

- Reactor Power is 100%
- Outside air temperature is 95° F
- "1B" Drywell Chiller is in service
- "1A" and "1B" Drywell Chilled Water Pumps are in service

A Dead Bus Transfer of the D12 Bus results in a loss of Drywell Chilled Water with the following indications:

- Drywell temperature is 143 ° F and up slow
- Drywell pressure rises to 0.7 psig

WHICH ONE of the following identifies the required action(s) to restore a DWCW flow path to the containment, if any?

A. No action required, flow path is maintained

B. Reopen the DWCW Containment isolation valves

C. Reset isolation R2 with Blue/Green reset per GP 8.3 and reopen the DWCW Containment isolation valves

D. Bypass the isolation per GP 8.5 and reopen the DWCW Containment isolation valves

Answer: B

Answer Explanation

From E-D12

NOTE

Loss of 10-Y102 (D124-R-G-23) will de-energize interposing relays
AND isolate Drywell Chilled Water (DWCW) Loop A
AND B isolation valves. The valves are powered from D124-R-C and will not isolate until power is restored to MCC. [ ]

Justification:

A. **Incorrect but plausible:** Plausible if the applicant fails to recognize that DW cooling isolation valves will close upon restoration of power to the MOVs due to the affects of the interposing relays.

B. **Correct:** The Dead Bus transfer of D12 will de-energize the interposing relays, resulting in isolation/closure of the DWCW Containment Isolation Valves upon re-energization of the bus. The valves can be re-opened when D12 power is restored because no isolation signal is present.

C. **Incorrect but plausible:** Plausible if the applicant believes an isolation signal exists. Performing an R2 with Blue/Green reset would clear an existing isolation signal when the monitored parameter (i.e., High DW Pressure for DWCW Containment Isolation Valves) has returned to a normal value. With no isolation signal present, the DWCW Containment Isolation Valves can be re-opened once power is restored to the D12 bus.

D. **Incorrect but plausible:** Bypassing the isolation per GP-8.5 is plausible if the applicant believes that an isolation signal exists. With no isolation signal present, the DWCW Containment Isolation Valves can be re-opened once power is restored to the D12 bus. In addition, GP-8.5 is not directed until DW temperature rises above 145°F in accordance with T-102, Step DW/T-4.

Question 46 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1799276
User-Defined ID:	Q #46
Lesson Plan Objective:	LGSOPS0072
Topic:	Loss of AC power effect on DWCW system
RO Importance:	3.7
SRO Importance:	3.7
K/A Number:	295003 AK3.06

Comments:	

General Data	
Level	RO
Tier	1
Group	1
KA # and Rating	295003
KA Statement	**Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER :** Containment isolation
Cognitive level	Higher
Safety Function	6
10 CFR 55	CFR: 41.5 / 45.6
Technical Reference with Revision No:	E-D12 **Rev #:** 11
Justification for Non SRO CFR Link:	
Question History: (i.e. LGS NRC-05, OYS CERT-04)	NRC LGS2012
Question Source: (i.e. New, Bank, Modified)	Bank 1097448
Low KA Justification (if required):	
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)	re-worded question and changed disctractor A due to the chiller and circulating pumps tripping when this is run in the simulator (low flow condition)
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	None
LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

Original Q#1097448

47	ID: 1799329	Points: 1.00

Unit 1 is in OPCON 3, with the following:

- MSIVs are closed
- Cooldown is in progress using SRVs
- HV-59-129A and HV-59-129B, 'PCIG PROCESS LINES', to the Drywell are closed due to loss of power

The PRO opens HV-59-128A and HV-59-128B, 'INST AIR TO INST GAS.'

WHICH ONE of the following describes the availability of the '1E' and '1F' SRVs for continuing the cooldown?

	'1E' SRV	'1F' SRV
A.	available	available
B.	available	NOT available
C.	NOT available	available
D.	NOT available	NOT available

Answer: B

Answer Explanation

From the stem the candidate determines that Normal PCIG supply to the Drywell has been isolated via HV-59-129A and HV-59-129B. These valves isolate the PCIG compressors and the PCIG Receivers. This is the normal Instrument Air/Gas supply to all SRV solenoid valves. With the opening of HV-59-128A and HV-59-128B Instrument Air is introduced to the PCIG header. This connection is upstream of the closed HV-59-129A and HV-59-129B, therefore Instrument Air will not enter the drywell to support Drywell loads. With the normal PCIG supply gone, the remaining Instrument Air/Gas supply is limited to N2 bottles (1AS252 and 1BS252) and PCIG thrugh HV-59-151A and HV-59-151B supplying only the ADS SRVs (S,H,M,E,K).

A Wrong - plausible to the candidate that fails to recall where the Instrument Air system backup connects to the PCIG and believes that the connection is down stream of the HV-59-129A and HV-59-129B and that Instrument Air is supplying all SRVs.
B Correct for the above reasons
C Wrong - plausible to the candidate the confuses the SRVs that receive backup N2 and PCIG supply
D Wrong - plausible to the candidate that fails to recall that ADS SRVs (S,H,M,E,K) have backup bottles and PCIG supplying N2 for operation

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:	0.00
System ID:	1799329
User-Defined ID:	Q #47
Lesson Plan Objective:	LGSOPS0059.3A
Topic:	Loss of IA/PCIG - Backup
RO Importance:	3.3
SRO Importance:	
K/A Number:	295019 AK3.01

Comments:

General Data	
Level	RO
Tier	1
Group	1
KA # and Rating	295019 AK3.01 RO Importance 3.3
KA Statement	295019 Partial or Total Loss of Inst. Air / 8 AK3.01 - Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Backup air system supply: Plant-Specific
Cognitive level	High
Safety Function	8 - Plant Service Systems
10 CFR 55	41.5
Technical Reference with Revision No:	M-0059, Sheet 001 M-0059, Sheet 002 — **Rev #:** 38 37
Justification for Non SRO CFR Link:	N/A
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Bank 556930
Question Source: (i.e. New, Bank, Modified)	Bank 556930
Low KA Justification (if required):	N/A
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)	
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	None
LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

48 **ID: 1799332** **Points: 1.00**

Plant conditions are as follows:

- PJM has declared a 'Maximum Emergency Generation Action'
- 230K Grid Voltage has dropped 10KV

As a result:

- Generator H_2 pressure is 75 psig
- Generator MW load is 1170 MWe
- Generator MVAR load is 481 MVARs
- Generator Field Current is 6,382 Amps
- Generator Terminal Voltage is 20.9 KV

MAIN GENERATOR CAPABILITY CURVE AND TABLE

GENERATOR HYDROGEN MACHINE GAS PRESSURE (PSIG)		75	74	73	72	71	70	69	68	67	66	65	64	63	62	61	60	59	58	57	56	55	54	53
	1245	224	180	122	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no
	1240	250	212	166	100	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no
	1235	274	239	199	149	70	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no
	1230	296	264	228	186	131	0	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no
	1225	316	286	254	217	172	111	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no
	1220	334	307	277	243	204	157	86	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no
	1215	352	326	298	267	232	192	140	49	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no
	1210	369	344	318	289	257	221	178	121	no	no	no	no	no	no	no	no	no	no	no	no	no	no	no
Main	1205	385	361	336	309	280	247	209	163	96	no	no	no	no	no	no	no	no	no	no	no	no	no	no
Gen	1200	400	378	354	328	300	270	236	197	147	69	no	no	no	no	no	no	no	no	no	no	no	no	no
Load	1195	415	393	370	346	320	291	260	225	183	130	0	no	no	no	no	no	no	no	no	no	no	no	no
(Mwe)	1190	429	408	386	363	338	311	282	250	213	169	109	no	no	no	no	no	no	no	no	no	no	no	no
	1185	443	422	401	379	355	330	302	273	240	201	154	84	no	no	no	no	no	no	no	no	no	no	no
	1180	456	436	415	394	371	347	321	294	263	229	189	138	49	no	no	no	no	no	no	no	no	no	no
	1175	469	449	429	409	387	364	339	313	285	253	218	175	119	no	no	no	no	no	no	no	no	no	no
	1170	481	462	443	423	402	379	356	331	305	276	243	206	161	97	no	no	no	no	no	no	no	no	no
	1165	493	475	456	436	416	395	372	348	323	296	266	233	194	145	68	no	no	no	no	no	no	no	no
	1160	505	487	468	449	430	409	387	365	341	315	287	256	222	181	128	0	no	no	no	no	no	no	no
	1155	516	499	481	462	443	423	402	380	357	333	307	278	246	210	167	108	65	no	no	no	no	no	no
	1150	527	510	492	474	456	436	416	395	373	350	325	298	269	236	198	152	115	0	no	no	no	no	no
	1145	538	521	504	486	468	449	430	409	388	366	342	317	289	259	226	186	155	100	no	no	no	no	no
	1140	545	535	510	490	475	460	438	425	395	375	350	320	292	260	230	220	185	150	no	no	no	no	no
	1135	560	555	515	510	485	475	450	430	410	385	358	330	310	275	240	235	212	175	0	no	no	no	no
	1130	570	563	530	520	495	487	468	445	400	405	380	350	320	295	255	265	235	195	150	0	no	no	no
	1125	568	565	550	535	510	496	475	465	437	425	390	368	335	315	272	285	265	225	175	105	no	no	no
	1120	572	570	555	540	530	510	485	468	455	430	410	380	355	332	300	315	280	255	210	135	0	no	no

WHICH ONE of the following describes the required operator action?

A. Notify the TSO that generator terminal voltage cannot be corrected without first reducing generator MW load.

B. Notify the TSO that generator terminal voltage cannot be corrected without first reducing the MVAR load.

C. Lower the setting on 90P-G103 to reduce the MVAR load and thus restore generator terminal voltage to above its lower limit.

D. Raise the setting on 90P-G103 to restore generator terminal voltage to above its lower limit.

Answer: A

Answer Explanation
S32.3.A sets limits for main generator parameters during heavy grid load. Step 4.6 and its NOTE requires maintaining generator field current **less than** 6382 amps. Stem conditions indicate generator field current is currently 6,382 amps; therefore, that limit is already being exceeded. That same step requires maintaining generator terminal voltage above 21.0 KV. Stem conditions indicate the generator terminal voltage is currently 20.9 KV; therefore, **it is too low**. E-5 Grid Emergency, step 3.4 requires maintaining generator parameters within the limits of the Generator Capability Curve or Table in Attachment 2. The Table shows that for a real load of 1170 MWe at 75 psig H2 pressure, the maximum allowed reactive load is 481 MVARs. [NOTE - Each MWe load versus MVAR load combination (at 75 psig) shown on the Table equates to operating the machine precisely at its 1265 MVA (apparent power) rating. Use the Power Triangle to demonstrate this claim: Square Root of [(1170 MW)^2 + (481 MVAR)^2] = 1265 MVA, where this 1265 MVA rating is shown on the Generator Capability Curve (page 2 of Attachment 2), as well as in the NOTE associated with step 3.1.21 of GP-5.]. 'A' is correct. In order to raise the generator terminal voltage, operators would have to RAISE the setting on the automatic voltage regulator adjuster, 90P-G103. However, with the MVAR load already at its max limit for a real load of 1170 MWe, raising the 90P-G103 setting will also cause MVARs to increase above the E-5 Attachment 2 Table limit (i.e., the machine's MVA rating would be exceeded). Additionally, the higher setting on 90P-G-103 would also increase the generator field current above its <6,382 amp limit. Therefore, operators should notify the TSO that generator terminal volts cannot be corrected without first reducing the generator's real load (MWe); that will allow for a higher MVAR load (to accomodate having to raise the 90P-G103 setting) without exceeding the machine's MVA rating, as well as provide more margin for the increased generator field current without exceeding the <6,382 amp limit. 'B' is wrong. To reduce the MVAR load operators must LOWER the setting on 90P-G103. The problem is, this will also lower the generator terminal voltage, making that situation worse than it already is. This is plausible to the examinee who does not completely understand the relationship between a 90P-G103 adjustment and the resulting generator terminal volts. 'C' is wrong. As explained for choice 'B' above, lowering the 90P-G103 setting will result in a lower generator terminal voltage, not a higher one. This is plausible to the examinee who does not completely understand the relationship between a 90P-G103 adjustment and the resulting generator terminal volts. 'D' is wrong. As already explained for the correct answer 'A', while restoring generator terminal volts to above the lower limit, raising the 90P-G103 setting will also raise the generator field current to above the 6,382 Amp limit. This is plausible to the examinee who either does not completely understand the relationship between a 90P-G103 adjustment and the resulting generator field current, or who fails to recognize that the pre-existing generator field current is already at (actually, is exceeding) its <6,382 amp limit.

Question 48 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1799332
User-Defined ID:	Q #48
Lesson Plan Objective:	LLOT1566.02
Topic:	E-5 - Determine Actions based on Main Gen Parameters During Grid Emergency
RO Importance:	3.6
SRO Importance:	3.7
K/A Number:	700000.AA1.01

Comments:

General Data		
Level	RO	
Tier	1	
Group	1	
KA # and Rating	700000 AA1.01 3.6 / 3.7	
KA Statement	**Ability to operate and/or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES:** Grid frequency and voltage	
Cognitive level	Higher	
Safety Function	6 - Electrical	
10 CFR 55	CFR: 41.5	
Technical Reference with Revision No:	S32.3.A E-5 GP-5	**Rev #:**
Justification for Non SRO CFR Link:		
Question History: (i.e. LGS NRC-05, OYS CERT-04)		
Question Source: (i.e. New, Bank, Modified)	Bank 988823	
Low KA Justification (if required):		
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)		
ILT		
Supplied Ref (If appropriate): (i.e. ABN-##)		
LORT		
PRA: (i.e. Yes or No or #)		
LORT Question Section: (i.e, A-Systems or B-Procedures)		

#988823

49 **ID: 1799353** **Points: 1.00**

Unit 1 is in OPCON 3, with the following:

- '1A' RHR Pump is in Shutdown Cooling (SDC)
- Drywell Pressure rises to 3 psig

Consider the following Shutdown Cooling Valves:

- HV-51-1F008 RHR S/D Clg Suction (OUTBOARD)
- HV-51-1F009 RHR S/D Clg Suction (INBOARD)
- HV-51-1F015A RHR S/D/ Clg Rtn (OUTBOARD)

WHICH ONE of the following identifies the Shutdown Cooling Valves status one minute after the LPCI initiation signal is generated and whether the 1A RHR Pump is adding inventory to the RPV?

(Assume no operator actions.)

	Shutdown Cooling Valves	1A RHR adding inventory to the RPV
A.	Open	No
B.	Open	Yes
C.	Closed	No
D.	Closed	Yes

Answer: A

Answer Explanation

From the stem the candidate is told that Shutdown Cooling is in service. This required RPV level above 12.5" and Reactor Pressure to be below 75# psig. When Drywell pressure rises above 1.68# with Reactor Pressure less than 455#, the conditions for a LOCA signal are met. As can be seen in the table below, Drywell pressure does not isolate Shutdown Cooling therefore the Shutdown Cooling Valves remain open.

From GP-8

Group	System	Signal	Reset Attach 3	Bypass Attach 4
IA	MSIV's and MSL Drains	C Rx Level -129" E MSL Hi Flow F TE MST Hi Temp P MSL Lo Press 756 psig Q Cond Vacuum Lo	R3 R3 R3 R3 R3	None None None None S25
IB	MS & Rx Sample	B Rx Level -38"	R2	None
IIA	RHR S/D Cooling	A Rx Level 12.5" V Rx Pressure 75 psig	R1 R1	None None

With Shutdown Cooling not isolated, and a LOCA signal, 1A RHR will inject however it will not be adding any inventory due to its suction source and discharge location both being the RPV.

A Correct for the above reasons
B Wrong - plausible to the candidate the fails to recall the suction source is the RPV and will not add inventory
C Wrong - plausible to the candidate that incorrectly believes that Shutdown cooling will isolate on 1.68 psig in the drywell
D Wrong - plausible to the candidate that incorrectly believes that Shutdown cooling will isolate on 1.68 psig in the drywell and believes the RHR suction is from the suppression pool such that 1A RHR would add inventory to the RPV

Question 49 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1799353
User-Defined ID:	Q #49
Lesson Plan Objective:	LGSOPS0051.8A
Topic:	High Drywell Pressure - Operate or monitor RHR/LPCI
RO Importance:	4.1
SRO Importance:	
K/A Number:	295024 EA1.04

Comments:

General Data			
Level	RO		
Tier	1		
Group	1		
KA # and Rating	295024 EA1.04 RO Importance 4.1		
KA Statement	295024 High Drywell Pressure / 5 EA1.04 - Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: RHR/LPCI		
Cognitive level	High		
Safety Function	5 - Containment Integrity		
10 CFR 55	41.7		
Technical Reference with Revision No:	GP-8	**Rev #:**	18
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Modified Bank 555993		
Question Source: (i.e. New, Bank, Modified)	Modified Bank 555993		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)	Changed LOCA signal to pressure and changed part 2 to adding inventory		
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

50	ID: 1799575	Points: 1.00

Plant conditions are as follows:

- Main Turbine tripped from 100% power
- 120 Control Rods failed to fully scram due to a hydraulic lock
- Reactor Power is 10%
- Reactor Pressure is 900 psig
- Reactor Mode Switch is in "SHUTDOWN"
- CRD discharge pressure is 1155 psig

WHICH ONE of the following identifies why the RO cannot manually insert the Control Rods?

A. RWM rod block

B. APRM rod block

C. CRD Drive Water Pressure control valve HV-46-1F003 failed closed

D. Full Scram Discharge Volume

Answer: A

Answer Explanation	
A	**Correct:** Reactor power is 10% which is below the LPSP. An out of sequence rod will have a Insert and Withdraw Blocks along with a ROD BLOCK annunciator. LPSP is approximately 13.9% Steam flow when power drops. To move control rods T-101 reminds the operator to bypass the RWM
B	**Wrong:** APRM Rod Block is Plausible to the candidate who incorrectly recalls the APRM rod block power level (12% when Mode switch is in Shutdown (out of Run position)) and believes that power is above the RBM power level. This is an out block however not an insertion block.
C	**Wrong:** Plausible to the candidate who confuses the operation of the Pressure Control Valve and believes that closing the valve would starve the flow control and drive water section of CRD when in actuality it would provide more pressure to the drive water header and assist in rod insertion.
D	**Wrong:** Plausible to the candidate who believes that CRD exhaust header discharges to the Scram discharge volume and therefore has no flowpath instead of CRD Exhaust Header.

Question 50 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1799575
User-Defined ID:	Q #50
Lesson Plan Objective:	LGSOPS1560.03
Topic:	Manually insert control rods with failure to scram due to a 120 rod hydraulic lock
RO Importance:	2.7
SRO Importance:	2.8
K/A Number:	295005 AA1.03

Comments:	

General Data	
Level	RO
Tier	1
Group	1
KA # and Rating	295005 AA1.03 2.7 / 2.8
KA Statement	**Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP :** Reactor manual control/Rod control and information system
Cognitive level	Higher
Safety Function	3 - Reactor Pressure Control
10 CFR 55	41.7
Technical Reference with Revision No:	**Rev #:**
Justification for Non SRO CFR Link:	
Question History: (i.e. LGS NRC-05, OYS CERT-04)	
Question Source: (i.e. New, Bank, Modified)	Bank 557084
Low KA Justification (if required):	
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	None
LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

51	ID: 1799355	Points: 1.00

A 30% ATWS is in progress on Unit 2.

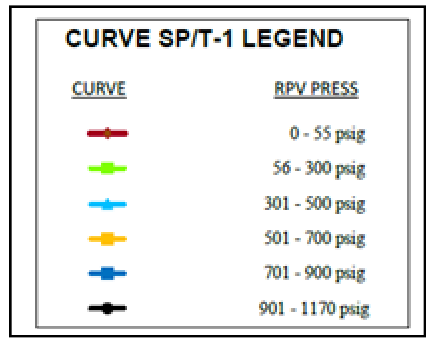
Consider the following 2 sets of conditions:

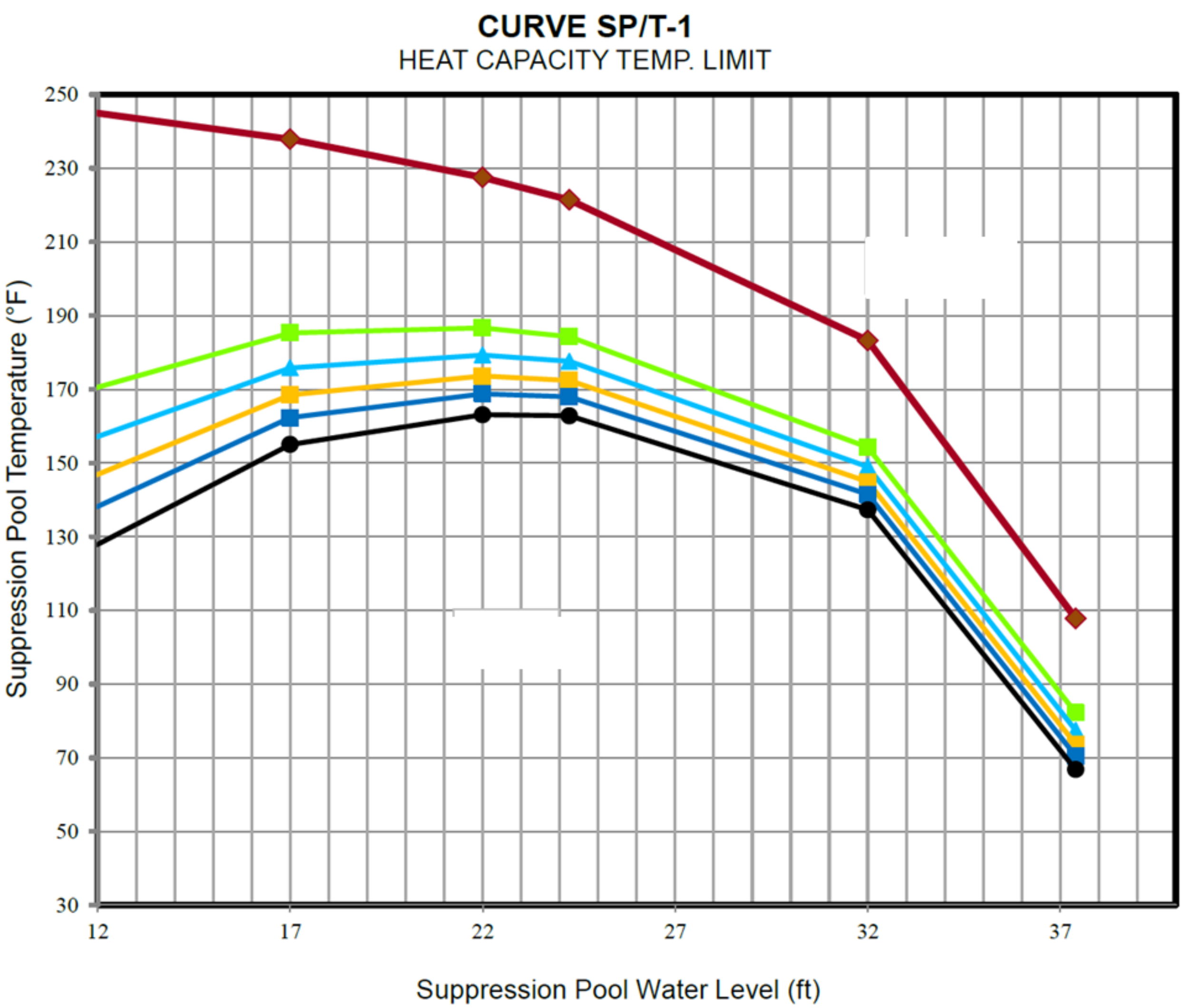
<u>Condition 1</u>

Suppression Pool level is 20' and stable
Suppression Pool temperature is 150 degrees and up slow
Reactor pressure is 1000 psig

<u>Condition 2</u>

Suppression Pool level is 23' and stable
Suppression Pool temperature is 155 degrees and up slow
Reactor pressure is 800 psig





WHICH ONE of the following (1) identifies the set of conditions which will provide the GREATEST margin to Blowdown if Reactor pressure remains constant and (2) defines Heat Capacity Temperature Limit (HCTL)?

A. (1) Condition 1
(2)The highest suppression pool temperature at which initiation of RPV depressurization will not result in exceeding the max temperature capability of the suppression pool

B. (1) Condition 1
(2) The suppression pool temperature limit that, when exceeded, requires RPV pressure reduction to the next lower pressure band to ensure the heat capacity of the suppression pool is not exceeded on a Blowdown

C. (1) Condition 2
(2)The highest suppression pool temperature at which initiation of RPV depressurization will not result in exceeding the max temperature capability of the suppression pool

D. (1) Condition 2
(2) The suppression pool temperature limit that, when exceeded, requires RPV pressure reduction to the next lower pressure band to ensure the heat capacity of the suppression pool is not exceeded on a Blowdown

Answer: C

Answer Explanation
The candidate must apply each of the conditions to the Curve from T-102. At 800 psig reactor pressure and 23 feet, HCTL will be exceeded at approx. 168. At 155 degrees, there is a 13 degree margin to blowdown. For condition 1, HCTL will be exceeded at 160, which means a margin of only 10 degrees (160-150), therefore LESS margin. From T-102 Bases, The HCTL curve is based on "the highest suppression pool temperature at which initiation of RPV depressurization will not result in exceeding the max temperature capability of the suppression pool". T-102 does not allow reducing reactor pressure once it is determined that the safe side of the HCTL curve can not be maintained. In this case, a Blowdown must be performed. This makes the last 2 choices incorrect. A Wrong - Plausible to the candidate that incorrectly applies the provided parameters to the curve B Wrong - Plausible to the candidate that incorrectly applies the provided parameters to the curve and fails to correctly recall the definition of HCTL C Correct for the above reasons D Wrong - Plausible to the candidate that fails to correctly recall the definition of HCTL

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:	0.00
System ID:	1799355
User-Defined ID:	Q #51
Lesson Plan Objective:	LGSOPS1560.04
Topic:	Suppression Pool high temp - impact Rx Pressure
RO Importance:	3.9
SRO Importance:	
K/A Number:	295026 EA2.03

Comments:	

General Data	
Level	RO
Tier	1
Group	1
KA # and Rating	295026 EA2.03 RO Importance 3.9
KA Statement	295026 Suppression Pool High Water Temp. / 5 EA2.03 - Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Reactor pressure
Cognitive level	High
Safety Function	5 - Containment Integrity
10 CFR 55	41.10
Technical Reference with Revision No:	T-102 Bases; **Rev #:** 26
Justification for Non SRO CFR Link:	N/A
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Bank 1149165
Question Source: (i.e. New, Bank, Modified)	Bank 1149165
Low KA Justification (if required):	N/A
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	Shaded in "Safe and "Unsafe" on graph
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	None
LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

Question 51 Table-Item Links

NRC EXAM - 10 CFR 55 Operators' Licenses

10 CFR 55.41 RO WRITTEN EXAMINATION

LIMERICK LO Question Category

ILT
NRC
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52	ID: 1799391	Points: 1.00

Plant conditions:

- A steam leak exists in the Drywell
- RPV pressure is 1040 psig, steady
- Drywell temperature is 200°F, up slow

WHICH ONE of the following describes the effect on RPV level indication for the above conditions?

A. Indicated level currently reads slightly lower than actual level

B. Indicated level currently reads slightly higher than actual level

C. Reference leg flashing will occur when drywell temperature reaches 212° F, causing indicated RPV level to rise

D. Indicated level is equal to actual level as long as drywell temperature remains less than 212°F

Answer: B

Answer Explanation

From T-102 Bases:

DW/T DRYWELL TEMPERATURE CONTROL

CAUTION
Elevated DW temp affects RPV level indication

DISCUSSION

This CAUTION warns operators of the possible effects of elevated drywell temperatures on RPV level instrument indications. This CAUTION, along with Step DW/T-1, define the conditions under which RPV level indications must be considered invalid due to the effects of RPV pressure, primary containment temperatures, and/or Reactor Enclosure temperatures.

RPV level instruments are calibrated to provide accurate indication under expected operating conditions. Indicated RPV level will be inaccurate if primary containment temperature, Reactor Enclosure temperature, or RPV pressure vary from the calibrated conditions of the associated instrument.

Note that the information provided in both this CAUTION and T-291, Temperature Effects On Reactor Level Instrumentation, do not simply correct for instrument inaccuracies due to variances from calibrated conditions. Rather, they define conditions under which neither the displayed value nor the indicated trend of RPV level can be relied upon.

A Wrong - plausible to the candidate that confuses the effect of temperature on the reference leg and reverses the relationship

B Correct - With drywell temperature elevated, reference leg water density will be lower, causing the d/p across the d/p cell to be lower, which causes indicated level to be higher than actual level.

C Wrong - plausable to the candidate that fails to recognize that the reference leg is pressurize and will not flash

D Wrong - plausable to the candidate that believes that the current elevated temperature has no effect on RPV level indication.

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:	0.00
System ID:	1799391
User-Defined ID:	Q #52
Lesson Plan Objective:	LGSOPS1560.4
Topic:	High Drywell Temperature - Determine and interpret Reactor Water Level
RO Importance:	3.7
SRO Importance:	3.9
K/A Number:	295028 EA2.03

Comments:	

General Data			
Level	RO		
Tier	1		
Group	1		
KA # and Rating	295028 EA2.03 RO Importance 3.7		
KA Statement	295028 High Drywell Temperature / 5 EA2.03 - Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : Reactor water level		
Cognitive level	High		
Safety Function	5 - Containment Integrity		
10 CFR 55	41.10		
Technical Reference with Revision No:	T-102 Bases	**Rev #:**	26
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	ILT Cert Exam 2005		
Question Source: (i.e. New, Bank, Modified)	ILT Cert Exam 2005 Bank 562330		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

Question 52 Table-Item Links

NRC EXAM - 10 CFR 55 Operators' Licenses

CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR 55.41 RO WRITTEN EXAMINATION

10 CFR 55.43 SRO WRITTEN EXAMINATION

CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

CFR: 45.6 Perform control manipulations required to obtain desired operating results during normal, abnormal, and emergency situations.

CFR: 45.8 Safely operate the facility's auxiliary and emergency systems, including operation of those controls associated with plant equipment that could affect reactivity or the release of radioactive materials to the environment.

CFR: 45.13 Demonstrate the applicant's ability to function within the control room team as appropriate to the assigned position, in such a way that the facility licensee's procedures are adhered to and that the limitations in its license and amendments are not violated.

53	ID: 1799469	Points: 1.00

Unit 2 plant conditions are as follows:

- Reactor Power is 20%, steady
- Drywell Pressure is 2.2 psig, steady
- RPV Level is -80", steady
- Suppression Pool temperature is 100 °F up 2.5 °F/min

Assume the above trend continues, WHICH ONE of the following identifies the number of minutes until the conditions above require the performance of T-270, Terminate and Prevent, and identifies the basis for this action?

	Minutes until conditions require performance of T-270	Basis for this Terminate and Prevent
A.	2	To prevent/mitigate the consequences of power oscillations
B.	2	Attempt to lower reactor power
C.	4	To prevent/mitigate the consequences of power oscillations
D.	4	Attempt to lower reactor power

Answer: D

Answer Explanation

From the stem the candidate identifies that three of the four conditions of T-117 step LQ-13; Reactor Power above 4%, RPV Level above -161 inches, and drywell pressure above 1.68 psig. The question asks when the conditions for Terminate and Prevent are met (in this case step LQ-13). The conditions will be met at a suppression pool temperature of 110 °F. At a rate of rise of 2.5 °F/min and a starting temperature of 100 °F this will take 4 minutes.

The basis for the Terminate and Prevent is found in T-117 bases for step LQ-13

DISCUSSION

LGS TRIP Step LQ-13 directs actions which attempt to lower reacto power by deliberately lowering RPV level when it has been determi that the primary containment is being threatened during a failure-to-scram event.

The second column distractor is taken from the basis for the first lowering, T-117 bases for step LQ-7.

To prevent or mitigate the consequences of any large irregular neutr flux oscillations induced by neutron flux/thermal-hydraulic instabilities, RPV level is lowered below -50 inches, which corresponds to an RPV level two feet below the elevation of the feedwater sparger nozzles. This places the feedwater spargers in th steam space, thereby providing effective heating of the relatively cold feedwater and eliminating the potential for high core inlet subcooling. For plant conditions that are susceptible to reactor power oscillations, the initiation and growth of these oscillations principally dependent upon the subcooling at the core inlet; the greater the subcooling, the more likely that reactor power oscillations will commence and rise in magnitude. For worst case conditions, terminating injection within 90 seconds of significantl reducing recirculation flow can mitigate oscillations before fuel overheating begins at 120 seconds.

A Wrong - plausible to the candidate that fails to recall the Suppression Pool Temperature at which Terminate and Prevent is required (105 °F can be easily confused due to it being the Tech Spec 3.6.2.1a.2.a temperature limit during testing which adds heat to the suppression chamber) and plausible to the candidate that confuses the basis for the first lowering (T-117 step LQ-7) with that of LQ-13.

B Wrong - plausible to the candidate that fails to recall the Suppression Pool Temperature at which Terminate and Prevent is required (105 °F can be easily confused due to it being the Tech Spec 3.6.2.1a.2.a temperature limit during testing which adds heat to the suppression chamber)

C Wrong - plausible to the candidate that confuses the basis for the first lowering (T-117 step LQ-7) with that of LQ-13.

D Correct for the above reasons

Question 53 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1799469
User-Defined ID:	Q #53
Lesson Plan Objective:	LGSOPS1560.03
Topic:	ATWS - determine/interpret Suppression Pool Temperature
RO Importance:	4.0
SRO Importance:	
K/A Number:	295037 EA2.04

Comments:

General Data			
Level	RO		
Tier	1		
Group	1		
KA # and Rating	295037 EA2.04 RO Importance 4.0		
KA Statement	295037 SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown / 1 EA2.04 - Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : Suppression pool temperature		
Cognitive level	High		
Safety Function	1- Reactivity Control		
10 CFR 55	41.10		
Technical Reference with Revision No:	T-117 Bases Tech Specs 3.6.2.1	**Rev #:**	19
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

54	ID: 1799491	Points: 1.00

Considering the Main Turbine Stop Valves input to RPS, WHICH ONE of the following identifies the Scram signal setpoint and describes the basis for this RPS trip during 100% power operation?

	Scram Signal Setpoint	Basis for RPS Trip
A.	Oil Pressure less than 500 psig	Anticipates the RPV pressure, neutron flux, and heat flux increase
B.	Oil Pressure less than 500 psig	Protect the turbine from excessive overspeed
C.	Valve Position less than 95% open	Anticipates the RPV pressure, neutron flux, and heat flux increase
D.	Valve Position less than 95% open	Protect the turbine from excessive overspeed

Answer: C

Answer Explanation

From Tech Spec Limiting Safety System Settings Bases page B 2-9:

9. Turbine Stop Valve-Closure

The turbine stop valve closure trip anticipates the pressure, neut flux, and heat flux increases that would result from closure of the sto valves. With a trip setting of 5% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins a maintained during the worst design basis transient.

Distractor from TRM Bases B3/4 3-7

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

This specification is provided to ensure that the turbine overspeed protection system instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine fi excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and dai safety related components, equipment or structures. This information in this TRM Section is also contained in UFSAR Section 10.2.2.6.1.

A Wrong - plausible to the candidate that mistakenly recalls the RPS set point for the Control Valves rather than the set point for the Stop Valves

B Wrong - plausible to the candidate that mistakenly recalls the RPS set point for the Control Valves rather than the set point for the Stop Valves and confuses the basis for overspeed protection with that of the RPS scram basis

C Correct for the above reasons

D Wrong - plausible to the candidate that confuses the basis for overspeed protection with that of the RPS scram basis

Question 54 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1799491
User-Defined ID:	Q #54
Lesson Plan Objective:	LGSOPS0071.04
Topic:	Stop Valve Scram setpoint and bases
RO Importance:	3.6
SRO Importance:	
K/A Number:	295006 G2.238

Comments:

General Data			
Level	RO		
Tier	1		
Group	1		
KA # and Rating	295006 G2.2.38 RO Importance 3.6		
KA Statement	295006 SCRAM / 1 2.2.38 - Equipment Control: Knowledge of conditions and limitations in the facility license.		
Cognitive level	Low		
Safety Function	1 - Reactivity Control		
10 CFR 55	41.7		
Technical Reference with Revision No:	Tech Specs	**Rev #:**	
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

55	**ID: 1801599**	**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

The Main Control Room receives:

- Alarm ARC-MCR-107 F4, SRM Period
- SRM '1C' count rate changing from 70 to 300 cps and continuing to rise.

WHICH ONE of the following describes the required actions?

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. Make PA announcement "Refuel Floor Secondary Containment breaches are to be restored per the Barrier Breach Contingency Plans."

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

Answer: D

Answer Explanation

From the stem the candidate determines the SRM count rates have more than doubled twice 70 - 140 - 280 CPS. This is symptom 1.1 of ON-120, Fuel Handling Problems". ON-120 section 1 addresses SRM count rates rising.

1. **IF** SRM count rate doubles two times between CCTAS steps **THEN PERFORM** the following:

   1.1 **IF** grappled, **RAISE** fuel assembly from core so it clears upper grid (approximately 360 inches hoist position) [

   1.2 **NOTIFY** Shift Supervision [

   1.3 **DETERMINE** SRM count rate trend [

   1.4 **IF** SRM count rate continues to increase (criticality) **THEN PERFORM** the following:

   - **EVACUATE** Fuel Floor [
   - **ENSURE** all insertable control rods are inserted [

   1.5 **NOTIFY** Health Physics and Reactor Engineering [

   1.6 **OBTAIN** permission from the Director, Operations before resuming fuel handling operations. [

A Wrong - Plausible to the candidate who does not recognize that the SRM count rate has more than doubled and has not stabilized and is increasing, indicating criticality. Determination of Dose rates is directed from a different section of ON-120 (section for any Fuel Floor ARM Alarm)

B Wrong - Plausible to the candidate 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 <u>no direction</u> to re-grapple.

C Wrong - Plausible to the candidate who recalls the actions for both - if an irradiated fuel bundle is dropped or damaged (ON-120 Attachment 3) and for if an Irradiated fuel rod is dropped or damaged (ON-120 Attachment 4) and believes this action is required for an SRM Rising event

D Correct for the above reasons

Question 55 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1801599
User-Defined ID:	Q #55
Lesson Plan Objective:	LGSOPS1550
Topic:	Recall ON-120 actions for signs of inadvertent criticality
RO Importance:	4.2
SRO Importance:	
K/A Number:	295023 G2.4.46

Comments:	

General Data	
Level	RO
Tier	1
Group	1
KA # and Rating	295023 G2.4.46 RO Importance 4.2
KA Statement	295023 Refueling Accident 2.4.46 - Emergency Procedures / Plan: Ability to verify that the alarms are consistent with the plant conditions.
Cognitive level	Low
Safety Function	8 - Plant Service Systems
10 CFR 55	41.10
Technical Reference with Revision No:	ON-120 **Rev #:** 28
Justification for Non SRO CFR Link:	N/A
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Bank 1149988 based on LGS NRC 2012 ILT (1097324)
Question Source: (i.e. New, Bank, Modified)	Bank 1149988
Low KA Justification (if required):	N/A
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)	Replaced distractor "C"
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	None
LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

Question 55 Table-Item Links

NRC EXAM - 10 CFR 55 Operators' Licenses

CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

CFR: 41.11 Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

10 CFR 55.41 RO WRITTEN EXAMINATION

LIMERICK LO Question Category

ILT

NRC

RO

LOW COG

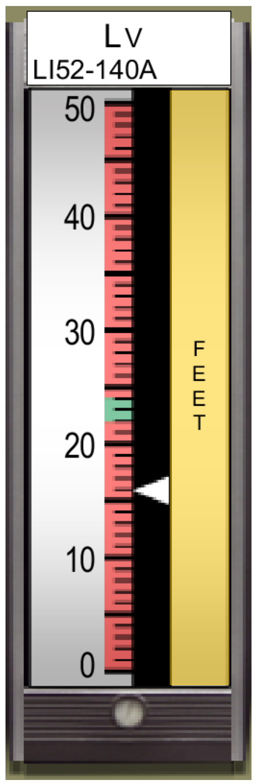
EXAMINATION ANSWER KEY

December 2017 ILT NRC - SRO Written

56	ID: 1799569	Points: 1.00

Plant conditions:

- Unit 1 Reactor SCRAMMED 20 minutes ago due to a group I isolation.
- Reactor pressure is steady at 970 psig and being maintained by HPCI in pressure control mode.
- Reactor level is steady at -20" and RCIC is injecting at 600 gpm.
- HPCI is running CST to CST at 5600 gpm
- Suppression Pool level is as indicated below:



WHICH ONE of the following describes the further operation of HPCI?

A. Can continue to operate at 5600 gpm

B. Can continue to operate but only at a reduced flowrate

C. HPCI must be secured only if suppression pool pressure begins to rise

D. HPCI must be secured

Answer: D

Answer Explanation

The candidate recognizes that because a Group 1 isolation occurred 20 minutes ago RCIC is/should be sufficient to maintain adequate core cooling by submergence.

Second, the candidate recalls the check/re-check from T-102 SP/L-4, which states that "If suppression Pool level cannot be maintained above 18 ft AND HPCI is not required from RPV injection, THEN secure HPCI."

This step is further defined from the basis as :

```
Operation of the HPCI turbine with its exhaust unsubmerged will tend
to directly pressurize the suppression chamber. If suppression pool
water level cannot be maintained above the elevation of the top of the
HPCI exhaust, HPCI is therefore secured if not needed for core
cooling.
```

Therefore, HPCI is secured because core cooling is assured.

A Wrong - plausible if the candidate fails to recall the fact that below a suppression Pool level of 18' HPCI exhaust will directly pressurize the Suppression Pool.
B Wrong - plausible to the candidate that confuses the provided conditions with those that would require reduced operation due to protecting from NPSH or suction vortexing.
C Wrong - plausible to the candidate that incorrectly applies the allowance of HPCI operation with the Suppression Pool Level below 18' for when HPCI is needed for adequate core cooling to this situation.
D Correct for the above reasons

Question 56 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1799569
User-Defined ID:	Q #56
Lesson Plan Objective:	LGSOPS1560.5
Topic:	T-102 - Determine actions required for HPCI with low Supp Pool level
RO Importance:	4.2
SRO Importance:	
K/A Number:	295030 G2.4.31

Comments:		
General Data		
Level	RO	
Tier	1	
Group	1	
KA # and Rating	295030 G2.4.31 RO Importance 4.2	
KA Statement	295030 Low Suppression Pool Water Level / 5 2.4.31 - Emergency Procedures / Plan: Knowledge of annunciator alarms, indications, or response procedures.	
Cognitive level	High	
Safety Function	5 - Containment Integrity	
10 CFR 55	41.2	
Technical Reference with Revision No:	T-102 Bases	**Rev #:** 26
Justification for Non SRO CFR Link:	N/A	
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Bank 1141088	
Question Source: (i.e. New, Bank, Modified)	Bank 1141088	
Low KA Justification (if required):	N/A	
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	Added picture of the Suppression Pool Level indicator.	
ILT		
Supplied Ref (If appropriate): (i.e. ABN-##)	None	
LORT		
PRA: (i.e. Yes or No or #)		
LORT Question Section: (i.e, A-Systems or B-Procedures)		
Comments		

Question 56 Table-Item Links

NRC EXAM - 10 CFR 55 Operators' Licenses

CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR 55.41 RO WRITTEN EXAMINATION

57	ID: 1799574	Points: 1.00

SRVs are automatically cycling open and closed due to high reactor pressure during an ATWS.

WHICH ONE of the following describes why SRV cycling is undesirable and warrants prompt manual action?

A. Prevent excessive damage to Bypass Valve/Main Condenser impingement plates during BPV operation at higher than normal RPV pressure

B. Minimize reactor power transients due to RPV pressure and level oscillations

C. Minimize SRV accumulator inventory loss

D. Minimize potential for a Downcomer Vacuum Breaker sticking open

Answer: B

Answer Explanation

LGS TRIP Step RC/P-7 is a decision diamond that has operators evaluate whether or not any of the SRVs are cycling.

"SRV cycling" is defined as multiple, closely sequenced SRV actuations where the SRV opens as RPV pressure exceeds the respective safety lift setpoint and closes as RPV pressure drops below the respective reset setpoint. SRV cycling is undesirable and warrants prompt manual action for the following reasons:

- It exerts significant dynamic loads upon the RPV, the SRV tail pipes and supporting structures, and the primary containment.
- Shrink and swell associated with SRV actuations cause RPV level fluctuations that complicate RPV level control actions.
- **Under failure-to-scram conditions, the consequent RPV level and RPV pressure oscillations can result in significant reactor power transients.**
- The potential for a stuck open SRV is increased.

A Wrong - plausible to the candidate the concludes that reactor pressure that causes SRV cycling (above 1170 psig) would cause significant damage on the impingment plates and this is the reasons for this step in T-101.

B Correct for the above reasons

C Wrong - plausible to the candidate the believes that SRVs cycling (on their lift pressure settings) uses gas from the accumulators (it does not)

D Wrong - plausible to the candidate that confuses Downcomer Vacuum Breakers (those that relieve pressure from the Suppression Pool air space to the Drywell air space) with SRV tail pipe vacuum breakers.

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:	0.00
System ID:	1799574
User-Defined ID:	Q #57
Lesson Plan Objective:	LGSOPS1560.5
Topic:	T-101 - Recall Basis for Step RC/P-7
RO Importance:	3.9
SRO Importance:	
K/A Number:	295025 EK2.09

Comments:

General Data			
Level	RO		
Tier	1		
Group	1		
KA # and Rating	295025 EK2.09 RO Importance 3.9		
KA Statement	295025 High Reactor Pressure / 3 EK2.09 - Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: Reactor power		
Cognitive level	Low		
Safety Function	3 - Reactor Pressure Control		
10 CFR 55	41.7		
Technical Reference with Revision No:	T-101 Bases	**Rev #:**	24
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Bank 560722		
Question Source: (i.e. New, Bank, Modified)	Bank 560722		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

Question 57 Table-Item Links

NRC EXAM - 10 CFR 55 Operators' Licenses

CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR 55.41 RO WRITTEN EXAMINATION

LIMERICK LO Question Category

ILT

NRC

RO

LOW COG

58	ID: 1845773	Points: 1.00

Plant conditions:

- A fire in the Cable Spreading Room has caused a MCR evacuation
- All Immediate Operator Actions are complete
- All Remote Shutdown Transfer switches are in EMERGENCY

WHICH ONE of the following interlocks remains **enabled**?

A. RCIC High Level Trip

B. D11, D12, D13 DG Breaker auto-closure

C. ESW Return to Spray Pond on '0A' ESW Pump start

D. HV51-1F016A ,'1A' Containment Spray Outboard Isolation Valve open permissive interlock

Answer: B

Answer Explanation

Answer: D11, D12, D13 DG Breaker auto close

SYSTEM INTERLOCK CHANGES DURING RSP OPERATIONS

RCIC

- Will not auto start.
- Will not automatically isolate.
-Will not trip on Reactor high level, (i.e. HV-50-*F045 will not close).
-Min Flow Valve operation is not automatic.
-Turbine Steam Inlet Valve is not interlocked to prevent opening with Turbine Exhaust Valve closed.
- Condensate Pump will not cycle automatically on high level.

SAFEGUARD BUSES
Interlocks inserted to prevent any breaker closure unless the other two breakers are open.

- D*1, D*2, AND D*3 D/G Breaker Control goes to "Auto Close" only.

RHR

-'A' LPCI auto start is defeated.
-'A' Min Flow Valve operation is not automatic.
- HV-51-*F017A, "'A' RHR Outboard Injection Valve," is not interlocked to prevent opening.
- HV-51-*F016A, "'A' Containment Spray Outboard Isolation Valve," is not interlocked to prevent opening.
- Group II Isolations for 'A' RHR Shutdown Cooling interlocks are defeated.
  (Reactor Level 3-Low, 12.5" AND Reactor Pressure-High, RHR Valve permissive, 75#)

RHRSW/ESW

-Automatic actions returning ESW to the Spray Pond on ESW 'A' OR 'C' Pump starts do not occur.
- RHRSW Pump does not trip on RHRSW Loop 'A' Return High Radiation OR High Discharge Pressure.

Distracters:

RCIC High Level Trip - see explanation of RSP interlocks

ESW Return to Spray Pond on "0A" ESW Pump start - see explanation of RSP interlocks

HV-51-1F016A "1A" Containment Spray Outboard Isolation Valve" open permissive interlock - see explanation of RSP interlocks

Question 58 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1845773
User-Defined ID:	Q #58
Lesson Plan Objective:	LLOT1563.03
Topic:	SE-1 - Recall Enabled Interlocks with RSP Operations
RO Importance:	3.1
SRO Importance:	3.2
K/A Number:	295016 AA1.04

Comments:	General Data			
	Level	RO		
	Tier	1		
	Group	1		
	KA # and Rating	295016 AA1.04 3.1 / 3.2		
	KA Statement	**Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT:** A.C. electrical distribution		
	Cognitive level	Lower		
	Safety Function	7 - Instrumentation		
	10 CFR 55	41.7		
	Technical Reference with Revision No:	SE-1	**Rev #:**	
	Justification for Non SRO CFR Link:			
	Question History: (i.e. LGS NRC-05, OYS CERT-04)	LGS		
	Question Source: (i.e. New, Bank, Modified)	Bank 560903		
	Low KA Justification (if required):			
	Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
	ILT			
	Supplied Ref (If appropriate): (i.e. ABN-##)	None		
	LORT			
	PRA: (i.e. Yes or No or #)			
	LORT Question Section: (i.e, A-Systems or B-Procedures)			
	Comments			

59 **ID: 1845763** **Points: 1.00**

The following sequence of events occurs on Unit 2:

- 10:00 REACTOR ENCL HVAC PANEL 2AC208 TROUBLE alarms
- 10:05 DIV 1 STEAM LEAK DET SYS HI TEMP/TROUBLE alarms
- 10:25 REACTOR ENCL ST FLOOD DAMPER PNL 20C234 TROUBLE alarms
- 10:40 EO reports 2AC208 TROUBLE alarm due to RCIC room temp 120 °F.

WHICH ONE of the following identifies when Initial and Re-Entry times into T-103, Secondary Containment Control, are required?

	Initial Entry Time	Re-Entry Time
A.	10:00	10:25
B.	10:00	10:40
C.	10:05	10:25
D.	10:05	10:40

Answer: D

Answer Explanation

Of the three alarms listed, Only the "DIV 1 STEAM LEAK DET SYS HI TEMP/TROUBLE" alarm required direct entry into T-103. From ARC-MCR-207 F5:

OPERATOR ACTIONS:

1. Enter T-103, Secondary Containment Control.

The other 2 alarms require investigation/confirmation prior to T-103 entry.
From ARC-MCR-219 A2:

OPERATOR ACTIONS:

1. Dispatch Operator to 2AC208 to determine which room cooler is causing the annunciator.
2. IF both room coolers are **not** operating in alarming room, THEN have Operator attempt to start coolers by placing both handswitches in RUN.
3. IF alarming condition is due to loss of Service Water, (check for proper Service Water Pump operation AND proper valve alignment), THEN initiate Emergency Service Water.
4. IF any ECCS pump Room is over 114 deg. F, THEN enter T-103, Secondary Containment Control.

From ARC-MCR-219 I3:

OPERATOR ACTIONS:

1. Dispatch Operator to panels 20C234 AND 20C245.
2. IF Steam Flooding Damper actuated, THEN enter T-103.

A Wrong - plausible to the candidate the confuses which alarms need confirmed prior to T-103 entry and which alarms require direct T-103 entry.
B Wrong - plausible to the candidate the confuses which alarms need confirmed prior to T-103 entry and which alarms require direct T-103 entry.
C Wrong - plausible to the candidate the confuses which alarms need confirmed prior to T-103 entry and which alarms require direct T-103 entry.
D Correct for the above reasons

Question 59 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1845763
User-Defined ID:	Q #59
Lesson Plan Objective:	LGSOPS1560.02A
Topic:	T-103, Identify the initial and reentry times for various T-103 conditions
RO Importance:	3.5
SRO Importance:	
K/A Number:	295032 EK1.03

Comments:

General Data			
Level	RO		
Tier	1		
Group	2		
KA # and Rating	295032 EK1.03 RO Importance 3.5		
KA Statement	295032 High Secondary Containment Area Temperature / 5 EK1.03 - Knowledge of the operational implications of the following concepts as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Secondary containment leakage detection: Plant-Specific		
Cognitive level	Low		
Safety Function	5 - Containment Integrity		
10 CFR 55	41.8		
Technical Reference with Revision No:	T-103 Bases ARC-MCR-219 A2 ARC-MCR-207 F5 ARC-MCR-219 I3	**Rev #:**	2 5 0 1 0 5 0 1
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Bank 561057		
Question Source: (i.e. New, Bank, Modified)	Bank 561057		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	Removed 10:30 MDDFP start from Stem and re-structured second column		
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

Question 59 Table-Item Links

NRC EXAM - 10 CFR 55 Operators' Licenses

10 CFR 55.41 RO WRITTEN EXAMINATION

CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

LIMERICK LO Question Category

ILT

NRC

RO

LOW COG

60	ID: 1845770	Points: 1.00

Unit 1 plant conditions:

- T-103, Secondary Containment Control, is in progress due to flooding in the RCIC Room
- The Reactor Enclosure Floor Drain Sump Pumps have tripped on Hi-Hi radiation
- The CRS has directed a procedure that places the HSS-61-104 REACTOR ENCL. FLOOR DRAIN SUMP SELECTION SW in the "HI-HI Rad" position.

WHICH ONE of the following correctly completes the following statements?

The purpose of the Hi-Hi Radiation trip of the Reactor Enclosure Sump Pump is to prevent transfer of highly radioactive water to the _____(1)_____. With HSS-61-104 selected to the HI-HI Rad position, the Reactor Enclosure Sump Pumps _____(2)_____ start/stop on sump level.

	(1)	(2)
A.	Suppression Pool	do not
B.	Suppression Pool	auto
C.	Radwaste Enclosure	do not
D.	Radwaste Enclosure	auto

Answer: D

Answer Explanation

From T-236:

T-236, Rev. 15
Page 4 of 7
RCB:plr

*** * UNIT 1 ONLY * ***

INITIALS

NOTE	
NORM:	Pumps auto start/stop on level **AND** trip on Hi-Hi radiation.
HI-HI RAD:	Pumps auto start/stop on level **AND** do **not** trip on Hi-Hi radiation.
RUN A:	A pump runs continuously - **no** trips except on thermal. B pump auto starts/stops on level. B pump will trip on Hi-Hi radiation.
RUN B:	B pump runs continuously - **no** trips except on thermal. A pump auto starts/stops on level. A pump will trip on Hi-Hi radiation.

4.2 **INSERT/ROTATE** 1424A key **AND POSITION** HSS-61-104, "Rx Encl Floor Drain Sump Pumps Selector Switch," at 10C452 (162-A8-180) (ATTACHMENT 1) to "HI-HI RAD " to defeat Sump Pump high radiation trip interlock (from Post-LOCA Monitors RIX-26-191A,B,C,D). ________

From T-103 Bases

SCC/RAD-7 Consider pumping RE floor drain sump to Supp Pool per T-236

DISCUSSION

LGS TRIP Step SCC/RAD-7 directs the actions required to eliminate the potential radioactivity release path from the Reactor Enclosure to the Radwaste Enclosure, and outside through the Radwaste Enclosure HVAC System.

T-236, Transferring Reactor Enclosure Floor Drain Sump To Suppression Pool Via Core Spray System, specifies the actions required to pump the contents of the Reactor Enclosure Floor Drain Sump to the suppression pool via the Core Spray pump suction piping. This action is taken to prevent the transfer of highly radioactive water to the Radwaste Enclosure during an accident, returning it instead to the primary containment, which is better equipped to contain the radioactive material.

Step SCC/RAD-7 is purposely written to make no demands or suggestions. The intent of Step SCC/RAD-7 is for operators to consider, and take the specified action as required, only if it will provide a benefit in eliminating the offsite release of radioactivity from the Reactor Enclosure to the Radwaste Enclosure, and outside through the Radwaste HVAC System.

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:	0.00
System ID:	1845770
User-Defined ID:	Q #60
Lesson Plan Objective:	LGSOPS2003.2
Topic:	T-236, Describe the purpose of performing T-236, Transferring Reactor Enclosure Floor
RO Importance:	3.1
SRO Importance:	
K/A Number:	295036 EK2.01

Comments:

General Data			
Level	RO		
Tier	1		
Group	2		
KA # and Rating	295036 EK2.01 RO Importance 3.1		
KA Statement	295036 Secondary Containment High Sump/Area Water Level / 5 EK2.01 - Knowledge of the interrelations between SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL and the following: Secondary containment equipment and floor drain system		
Cognitive level	Low		
Safety Function	5 - Containment Integrity		
10 CFR 55	41.7		
Technical Reference with Revision No:	T-236 T-103 Bases	**Rev #:**	15 25
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

Question 60 Table-Item Links

NRC EXAM - 10 CFR 55 Operators' Licenses

CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

CFR: 41.6 Design, components, and functions of reactivity control mechanisms and instrumentation.

CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

CFR: 41.8 Components, capacity, and functions of emergency systems.

CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR 55.41 RO WRITTEN EXAMINATION

10 CFR 55.43 SRO WRITTEN EXAMINATION

CFR: 45.6 Perform control manipulations required to obtain desired operating results during normal, abnormal, and emergency situations.

CFR: 45.8 Safely operate the facility's auxiliary and emergency systems, including operation of those controls associated with plant equipment that could affect reactivity or the release of radioactive materials to the environment.

CFR: 45.13 Demonstrate the applicant's ability to function within the control room team as appropriate to the assigned position, in such a way that the facility licensee's procedures are adhered to and that the limitations in its license and amendments are not violated.

61	ID: 1845774	Points: 1.00

Unit 2 plant conditions:

- 50% power
- 2A RFP is blocked for maintenance
- Both Recirc Pumps running at 50% speed
- 2B RFP min flow valve fails open in AUTO
- Reactor level drops to + 25 inches before recovering

WHICH ONE of the following describes the response of the Recirc Pumps and the reason for this response?

A. Runback to 28% speed; reduces power to restore level by reducing steam load

B. Runback to 42% speed; reduces power to prevent RFP trip on low suction pressure

C. Runback to 28% speed; reduces power to ensure adequate NPSH to Recirc Pumps

D. Runback to 42% speed; reduces power to restore level by reducing steam load

Answer: D

Answer Explanation

From the stem the candidate identifies that flow through the 2A Reactor Feed Pump is less than 0.94 Mlbm/hr and RPV water level is less than 27.5 inches.

This satisfies the conditions required for a High Limit Runback (42% Speed)

From S43.0.B

1. **IF** any of the following conditions exist,
   THEN HIGH LIMIT RUNBACK will occur:

   - Any reactor feed pump flow less than 0.94 Mlbm/hour
     AND RPV water level less than 27.5 inches

     OR

   - Total feedwater flow greater than 12.0 Mlbm/hour
     AND any condensate pump **not** running

     OR

   - Loss of Stator Water Cooling

A Wrong - plausible to the candidte that incorrectly recalls which runback is received for the stem conditions and chooses 28%

B Wrong - plausible to the candated that incorrectly recalls the basis for the 42% runback upon a trip of a condensate pump and applies it to the stem conditions.

C Wrong - plausible to the candidte that incorrectly recalls which runback is received for the stem conditions and chooses 28% and plausible to the candated that incorrectly recalls the basis for the 28% runback.

D Correct, per S43.0.B, NOTE Section 4.1; the basis for the 42% runback on FW loop low flow and level <27.5" is to reduce reactor power to restore level

Question 61 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1845774
User-Defined ID:	Q #61
Lesson Plan Objective:	LLOT0040.04
Topic:	RPV low level - Recall Recirc Pump response and reason for
RO Importance:	3,2
SRO Importance:	3.3
K/A Number:	295009 AK3.01

Comments:	

General Data			
Level	RO		
Tier	1		
Group	2 - Reactor Water Inventory Control		
KA # and Rating	295009 AK3.01 3.2 / 3.3		
KA Statement	**Knowledge of the reasons for the following responses as they apply to LOW REACTOR WATER LEVEL :** Recirculation pump run back: Plant-Specific		
Cognitive level	Lower		
Safety Function	2		
10 CFR 55	41.5		
Technical Reference with Revision No:	S43.0.B	**Rev #:**	27
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	2012 CERT		
Question Source: (i.e. New, Bank, Modified)	Bank 1148918		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			

ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	None

LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

62	ID: 1845775	Points: 1.00

Unit 1 is at 95% power with the following conditions:

- Outside air temperature is 89 °F
- Circulating water temperature is 90 °F.

When the 1C Circ Water Pump trips.

WHICH ONE of the following identifies the predicted condensate temperature and procedure entry required as a result of the 1C Circ Water Pump trip?

	Predicted Condensate Temperature	Procedure Entry Required
A.	139 °F	OT-104
B.	139 °F	OT-116
C.	141 °F	OT-104
D.	141 °F	OT-116

Answer: B

Answer Explanation

From the stem the candidate determines there is a reduction in circ water flow during a hot day with Unit 1 operating at 95% power. Using this information and S09.2.A Attachment 1 Page Table 1, the predicted Condensate Temperature with Circ water temp of 90 °F and 95% power operation is 139 °F.

Table 1: Removing a Circulating Water pum

Circulating water temp. (Deg F.)	100%	95%	90%
75	125		
76	126		
77	127		
78	128		
79	129		
80	130	128	
81	131	130	
82	132	131	128
83	133	132	129
84	134	133	130
85	135	134	131
86	136	134	132
87	138	135	133
88	139	136	134
89	140	137	135
90	141	139	136
91	142	140	137
92	143	141	138
93	144	142	139
94	145	143	140
95	146	144	141

Under the given conditions a reduction in Main condenser vacuum would result requiring entry into OT-116, Loss of Condenser Vacuum.

A Wrong - plausible to the candidate that believes that a trip of a Circ Water pump would have a direct impact on core reactivity and cause a reactor power change.

B Correct for the above reasons

C Wrong - plausible to the candidate that applies the given plant conditions to Attachment 1 but reverses power and circ water temperature and plausible to the candidate that believes that a trip of a Circ Water pump would have a direct impact on core reactivity and cause a reactor power change.

D Wrong - plausible to the candidate that applies the given plant conditions to Attachment 1 but reverses power and circ water temperature.

Question 62 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1845775
User-Defined ID:	Q #62
Lesson Plan Objective:	LGSOPS1450 2
Topic:	Loss of Vacuum - Circ Water
RO Importance:	3.1
SRO Importance:	
K/A Number:	295002 AA1.07

Comments:

General Data	
Level	RO
Tier	1
Group	2
KA # and Rating	295002 AA1.07 RO Importance 3.1
KA Statement	295002 Loss of Main Condenser Vac / 3 AA1.07 - Ability to operate and/or monitor the following as they apply to LOSS OF MAIN CONDENSER VACUUM : Condenser circulating water system
Cognitive level	High
Safety Function	3 - Reactor Pressure Control
10 CFR 55	41.7
Technical Reference with Revision No:	S09.2.A — **Rev #:** 25
Justification for Non SRO CFR Link:	N/A
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New
Question Source: (i.e. New, Bank, Modified)	New
Low KA Justification (if required):	N/A
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	S09.2.A Attachment 1 (Pages 6 & 7)
LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

63	ID: 1845777	Points: 1.00

Unit 1 is operating at 100% power, normal operating level

- A DFWLC malfunction causes water level to rise at a rate of 12 inches per minute.
- The feedpumps and main turbine fail to trip on high level
- The Reactor Operator trips the feedpumps 3 minutes later (3 minutes after malfunction) and scrams the reactor

What level instrument will provide the most accurate indication of Reactor Water Level at the time the feedpumps are tripped?

A. Narrow Range

B. Shutdown Range

C. Wide Range

D. Upset Range

Answer: D

Answer Explanation	
A	Wrong: Plausible to the candidate who believes that water level stops rising prior to exceeding +60" on Narrow range because they do the math incorrectly
B	Wrong: Plausible to the candidate who recalls that the range of Shutdown overlaps with Upset range but fails to recall that Shutdown range calibrated cold. The instruments are calibrated for a water temperature of 120°F at 0 psig in the RPV and 80°F in the Drywell
C	Wrong: Plausible to the candidate who believes that water level stops rising prior to exceeding +60" on Wide Range because they do the math incorrectly or believe that they can read level up to +70" and therefore must be correct
D	Correct: With the conditions given, Rx level at NOL of 35" and then level begins to rise at 12 inches per minute. when the pumps are tripped 3 minutes later, level would have risen to 71". This is above the range of the Narrow Range and PAMS indication which are limited to +60 inches. The Upset instruments are calibrated for saturated steam and water conditions at 1045 psig in the RPV and 117/135°F in the Drywell. The range of the upset instrument is 0 to +180 inches. Shutdown range indicates from 0 to +370 inches but as noted above, is calibrated for cold conditions. Therefore Upset is the most accurate indication of high level at Hot conditions.

Question 63 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1845777
User-Defined ID:	Q #63
Lesson Plan Objective:	LLOT0041A.1K
Topic:	High RPV Level
RO Importance:	3.9
SRO Importance:	3.9
K/A Number:	295008 AA2.01

Comments:

General Data			
Level	RO		
Tier	1		
Group	2		
KA # and Rating	295008 AA2.01 3.9 / 3.9		
KA Statement	**Ability to determine and/or interpret the following as they apply to HIGH REACTOR WATER LEVEL:** Reactor water level		
Cognitive level	Higher		
Safety Function	2		
10 CFR 55	41.10		
Technical Reference with Revision No:	LGSOPS0042 DBD L-S-16	**Rev #:**	2 7
Justification for Non SRO CFR Link:			
Question History: (i.e. LGS NRC-05, OYS CERT-04)			
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):			
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

64	ID: 1845791	Points: 1.00

Unit 1 is performing a startup:

Reactor Power is 4%
Control Rod 34-27 is being withdrawn to position 48.

Alarm 108 REACTOR F-5, ROD OVERTRAVEL is received.

WHICH ONE of the following identifies what postulated UFSAR Chapter 15 transient is of concern in this situation and a component that mitigates the effect of this transient?

	UFSAR Chapter 15 transient	Component that minimizes the transient
A.	Control Rod Drop Accident	Velocity Limiter
B.	Control Rod Drop Accident	Collet Fingers
C.	Control Rod Withdrawal Error	Velocity Limiter
D.	Control Rod Withdrawal Error	Collet Fingers

Answer: A

Answer Explanation

From the stem the candidate determines that Control Rod 34-27 is not coupled to the Control Rod Drive (CRD) and the control rod position is unknown (it could be anywhere between position 00 and 48). This is the situation of the Control Rod Drop Accident with the worst case being the control rod stuck at position 00 and dropping uncontrollably from 00 to 48. The analysis described in UFSAR 15.4.9.3.3 results in the assumed failure of 1200 fuel rods. This would result in radiation level requiring entry into The Emergency Action Plan, at a minimum for Threshold RU3 - 2 (Specific coolant activity > 4.0 micro curies per gram.) The equipment credited in minimizing the impact of this accident are the Rod Worth Minimizer (which enforces a control rod sequence that minimizes local power peaks) and the velocity limiter (physical component on the Control Rod Blade that creates hydraulic drag to limit the speed of the control rod and the subsequent rate of reactivity insertion).

A Correct for the above reasons

B Wrong - plausible if the candidate confuses the components of the CRD with the Control Rod Blade. The Collet Fingers are a component designed to prevent the CRD from withdrawing without a command signal, not the Control Rod Blade

C Wrong - Plausible to the candidate that believes the Control Rod Withdrawal error transient is the concern in this situation due to the assumption that the control rod has not actually been withdrawn (due to it being uncoupled).

D Wrong - Plausible to the candidate that believes the Control Rod Withdrawal error transient is the concern in this situation due to the assumption that the control rod has not actually been withdrawn (due to it being uncoupled) and plausible if the candidate confuses the components of the CRD with the Control Rod Blade. The Collet Fingers are a component designed to prevent the CRD from withdrawing without a command signal, not the Control Rod Blade

Question 64 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	0.00
System ID:	1845791
User-Defined ID:	Q #64
Lesson Plan Objective:	LGSOPS1550.2
Topic:	Reactivity Addition - Low Power Accident
RO Importance:	3.8
SRO Importance:	
K/A Number:	295014 G2.4.9

Comments:

General Data			
Level	RO		
Tier	1		
Group	2		
KA # and Rating	295014 G2.4.9 RO Importance 3.8		
KA Statement	295014 Inadvertent Reactivity Addition / 1 2.4.9 - Emergency Procedures / Plan: Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.		
Cognitive level	High		
Safety Function	1 - Reactivity Control		
10 CFR 55	41.10		
Technical Reference with Revision No:	UFSAR 15.4 Tech Spec Bases 3/4.1.1	**Rev #:**	0 8
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

65	ID: 1845792	Points: 1.00

Unit 2 is operating at 100% power

- 30 minutes ago the Div 1 DC Bus was lost
- The 2A MSL Flow Element Sensing Line Fails such that all the instruments on that line see upscale high flow.

2 minutes later

- Reactor Pressure is 1030 psig up slow
- Reactor Level is -48 inches up slow

Which ONE of the following is available to control Reactor Pressure?

A. Main Turbine Bypass Valves

B. Steam Spargers

C. HPCI CST to CST

D. ADS SRVs from AER

Answer: D

Answer Explanation

A Wrong: Plausible to the candidate who doesn't recognize that the failure of the MSL transmitter would cause a full Group I isolation and make the BPV inoperable

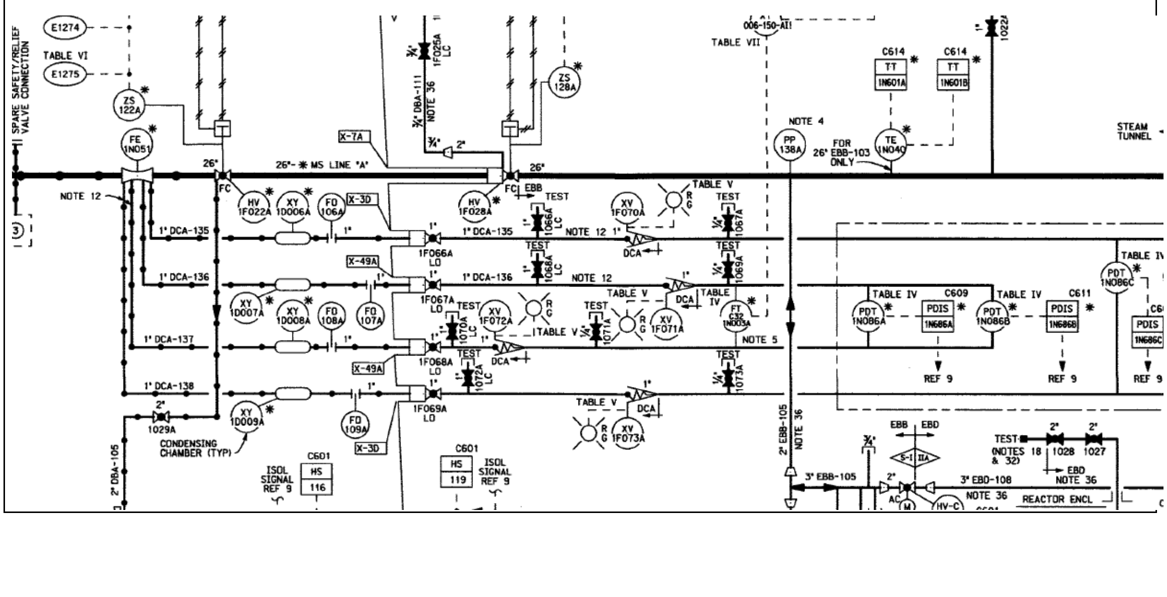
B Wrong: Plausible to the candidate who either doesn't recognize the Group I isolation or who believes that the Steam spargers would still be operable with the MSIVs closed

C Wrong: Plausible to the the candidate who does not consider that level is below -38" and therefore the HPCI initiation signal would prevent placing HPCI in Full Flow test as the F011 valve would automatically close with hi drywell pressure (.1.68#) or low Rx level (-38").

D Correct: Based on the explanation below for the MSL isolation logic, all MSIVs would receive an isolation signal. With all MSIVs closed, BPVs and Spargers are isolated from the condenser. HPCI is unavailable as noted above. The loss of Div 1 DC would appear to make ADS SRVs unavailable however, these valves can be locally operated from the AER panels or using ADS pushbuttons in the MCR, however this method is not desireable as all 5 ADS SRVs would open.

D/P transmitters (PDTs) (channels 'A', 'B', 'C', 'D') sense the d/p and convert it to a usable signal. Notice that the 4 PTDs are connected to the FE in pairs...e.g., PDTs 'C' & 'D' (a pair) share a common low-pressure sensing tap on the FE and a common high-pressure sensing tap on the FE. (Similar for the other pair, PDTs 'A' & 'B').

This question supposes that "a pressure sensing line on the flow restrictor has ruptured at the tap." Based on the information given in the step and the A Main Steam Flow element indicating off scale high, the failure is such that there is a false UPSCALE flow that is affecting the Flow Transmitter (FT-C32-1N003A in the section of the print provided). Once this is concluded, it can be observed that the A and B flow tranmitters are affected with an UPSCALE flow signal too. This is sufficient to satisfy the (**A** or C) AND (**B** or D) logic to cause **ALL 8 MSIVs to isolate** on high steam flow.



Question 65 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1845792
User-Defined ID:	Q #65
Lesson Plan Objective:	LLOT1566.02
Topic:	Loss of Heat Sink
RO Importance:	3.7
SRO Importance:	3.9
K/A Number:	295020 AK1.01

Comments:

General Data			
Level	RO		
Tier	1		
Group	2		
KA # and Rating	295020 AK1.01 3.7 / 3.9		
KA Statement	**Knowledge of the operational implications of the following concepts as they apply to INADVERTENT CONTAINMENT ISOLATION:** Loss of normal heat sink		
Cognitive level	Lower		
Safety Function	5, 7		
10 CFR 55	41.8		
Technical Reference with Revision No:	E-2FA	**Rev #:**	9
Justification for Non SRO CFR Link:			
Question History: (i.e. LGS NRC-05, OYS CERT-04)			
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):			
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

66	ID: 1845795	Points: 1.00

Plant conditions:

- Both Units are operating at 100% power
- The entire shift attended the morning Shift Turnover Meeting given by the CRS and Shift Manager

At 1100 hours, the PRO relieves the Unit 2 RO for lunch.

The Unit 2 RO will be eating lunch in the MCR Lunchroom.

WHICH ONE of the following identifies the required **MINIMUM** turnover activity and the required **MAXIMUM** duration of the PRO's mid-shift turnover, per OP-AA-112-101 (Shift Turnover and Relief)?

A. Read the MCR logs through the last previous day on shift.
The relief duration shall not exceed 30 minutes.

B. Read the MCR logs through the last previous day on shift.
The relief duration shall not exceed 60 minutes.

C. Review the Shift Turnover Checklist and review updated plant status.
The relief duration shall not exceed 30 minutes

D. Review the Shift Turnover Checklist and review updated plant status.
The relief duration shall not exceed 60 minutes.

Answer: D

Answer Explanation
Correct: Review the Shift Turnover Checklist and review updated plant status. The relief duration shall not exceed 60 minutes: Mid-shift turnover of less than 1 hour consists of a review of the Shift Turnover checklist and update of plant status, provided the original watch stander remains in the Main Control Room area **Incorrect: Read the MCR logs through the last previous day on shift. The relief duration shall not exceed 30 minutes:** Reading the MCR logs through the last previous day on shift applies to a full shift turnover, short turnover duration is 60 minutes, not 30. **Incorrect: Read the MCR logs through the last previous day on shift. The relief duration shall not exceed 60 minutes:** Duration is correct, turnover criteria described applies to full turnover, not short duration **Incorrect: Review the Shift Turnover Checklist and review updated plant status. The relief duration shall not exceed 30 minutes:** Turnover criteria is correct, short turnover duration is 60 minutes, not 30.

Question 66 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1845795
User-Defined ID:	Q #66
Lesson Plan Objective:	LGSOPS2010.24B
Topic:	OP-AA-112-101 - Recall Shift Turnover requirements
RO Importance:	3.7
SRO Importance:	3.9
K/A Number:	2.1.3

Comments:

General Data			
Level	RO		
Tier	3		
Group			
KA # and Rating	G 2.1.3 3.7 / 3.9		
KA Statement	**Knowledge of shift or short-term relief turnover practices.**		
Cognitive level	Lower		
Safety Function			
10 CFR 55	41.10		
Technical Reference with Revision No:	OP-AA-112-101	**Rev #:**	12
Justification for Non SRO CFR Link:			
Question History: (i.e. LGS NRC-05, OYS CERT-04)	NRC-05, LGS ILT07-1 CERT		
Question Source: (i.e. New, Bank, Modified)	Bank 591149		
Low KA Justification (if required):			
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

Previous Question #591149

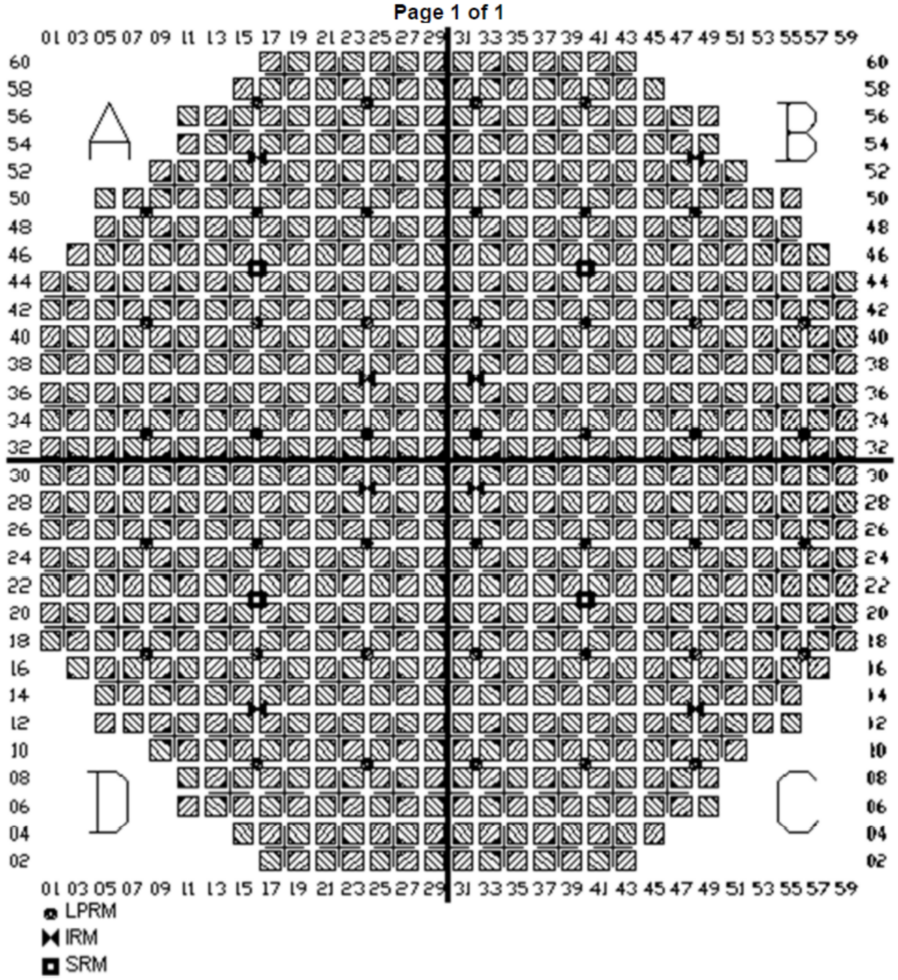
67	ID: 1845800	Points: 1.00

Unit 2 is in OPCON 5 with Core Shuffle 2 in progress.

- The 2A SRM is bypassed
- The 2D SRM is INOP due to spiking

WHICH ONE of the following identifies a core location where a fuel assembly may be inserted, if any, for the above conditions?

ATTACHMENT 1
SRM Quadrant Boundaries
Page 1 of 1



A. 35-06

B. 27-56

C. 07-18

D. No locations

Answer: A

Answer Explanation
From the stem the candidate determines that the 2A and 2D SRMs are INOP. Using this information, attachment 1 from NF-LG-310-2000 provided, and their knowledge of Tech Spec LCO 3.9.2 they determine that core alterations can continue in the B and C quadrants. Only 35-06 is in a quadrant where core alterations can continue.

REFUELING OPERATIONS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 At least two source range monitor (SRM) channels* shall be OPERABLE and inserted to the normal operating level with:

a. Continuous visual indication in the control room,

b. At least one with audible alarm in the control room,

c. One of the required SRM detectors located in the quadrant where ALTERATIONS are being performed and the other required SRM detec located in an adjacent quadrant, and

d. Unless adequate SHUTDOWN MARGIN has been demonstrated, the "shor links" shall be removed from the RPS circuitry prior to and duri the time any control rod is withdrawn.**

A Correct for the above reasons

B Wrong - plausible to the candidate that incorrectly uses attachment 1 (SRM Locations, Fuel Assembly coordinates, etc.)

C Wrong - plausible to the candidate that incorrectly uses attachment 1 (SRM Locations, Fuel Assembly coordinates, etc.)

D Wrong - plausible to the candidate the incorrectly applies Tech Spec 3.3.7.6.a where 3 SRMs are required to be operable for startup with IRMs on range 2 or below.

Question 67 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1845800
User-Defined ID:	Q #67
Lesson Plan Objective:	LGSOPS1800. 5A
Topic:	Refueling Administrative requirements
RO Importance:	2.8
SRO Importance:	
K/A Number:	G2.1.40

Comments:

General Data	
Level	RO
Tier	3
Group	
KA # and Rating	G2.1.40 RO Importance 2.8
KA Statement	G2.1.40 Knowledge of refueling administrative requirements
Cognitive level	High
Safety Function	N/A
10 CFR 55	41.10
Technical Reference with Revision No:	Tech Specs LCO 3.9.2 — **Rev #:**
Justification for Non SRO CFR Link:	N/A
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New
Question Source: (i.e. New, Bank, Modified)	New
Low KA Justification (if required):	N/A
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	None
LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

68	ID: 1845797	Points: 1.00

WHICH ONE of the following identifies the Unit 1 and Unit 2 RHR Loops used for alternate injection using RHRSW?

	Unit 1 RHR Loop	Unit 2 RHR Loop
A.	A	A
B.	A	B
C.	B	A
D.	B	B

Answer: C

Answer Explanation	
A	**Wrong:** Plausible to the candidate who does not recall the connections for RHRSW to Unit 1 and 2
B	**Wrong:** Plausible to the candidate who does not recall the connections for RHRSW to Unit 1 and 2
C	**Correct:** Unit 1 = "0B" RHRSW via "1B" Loop RHR Unit 2 = "0A" RHRSW via "2A" Loop RHR T-243 ALTERNATE INJECTION BY WAY OF RHRSW TO RHR LOOP "B" for Unit 1 and LOOP "A" for Unit 2
D	**Wrong:** Plausible to the candidate who does not recall the connections for RHRSW to Unit 1 and 2

Question 68 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1845797
User-Defined ID:	Q #68
Lesson Plan Objective:	LLOT1870.1
Topic:	Identify the Unit 1 and Unit 2 injection points for the alternate injection subsystems
RO Importance:	RO 3.1
SRO Importance:	SRO 3.3
K/A Number:	K/A 2.2.3

Comments:	General Data			
	Level	RO		
	Tier	3		
	Group			
	KA # and Rating	G 2.2.3		
	KA Statement	**Knowledge of the design, procedural, and operational differences between units.**		
	Cognitive level	Low		
	Safety Function			
	10 CFR 55	41.5		
	Technical Reference with Revision No:	T-243 U1/2	**Rev #:**	16/12
	Justification for Non SRO CFR Link:	N/A		
	Question History: (i.e. LGS NRC-05, OYS CERT-04)	Modified 561574 2002 NRC		
	Question Source: (i.e. New, Bank, Modified)	Modified 561574		
	Low KA Justification (if required):	N/A		
	Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	Changed question to address RHRSW only to avoid overlap with T-244 inplant JPM.		
	ILT			
	Supplied Ref (If appropriate): (i.e. ABN-##)	None		
	LORT			
	PRA: (i.e. Yes or No or #)			
	LORT Question Section: (i.e, A-Systems or B-Procedures)			
	Comments			

69	ID: 1845878	Points: 1.00

Unit 1 is operating at 100% power

- RCIC failed a surveillance test resulting in RCIC being declared INOPERABLE and UNAVAILABLE
- RCIC return to operable is expected to be 96 hours

WHICH ONE of the following correctly identifies (1) the location of required protected equipment postings for the HPCI system and (2) if access near the HPCI system while it is protected is permitted for an Operator performing rounds, inspections, and alarm response?

	Location of HPCI protection postings	Access for Operator permitted
A.	In the field ONLY	No
B.	In the field ONLY	Yes
C.	Main Control Room and in the field	No
D.	Main Control Room and in the field	Yes

Answer: D

Answer Explanation

From OP-AA-108-117

4.3.8. The following considerations and examples are to be evaluated when posting protected equipment:

– For extent of protection, barriers must extend back at least one component f protected equipment. For example:

- Concerning protected pumps and fans, the local and remote control swit the pump/fan general area, the power supply (i.e. back to the pump / fan feeder breaker), specific instruments (or instrument racks as appropriate could cause a pump/fan trip or are required for monitoring, and necessar support systems (e.g. cooling water).

4.4. Access or Work on or Near Protected Equipment **(CM-1)**

4.4.1. Generally, access or work on or near protected equipment will not be allov Exceptions to this rule are as follows:

– Operator performing rounds, inspections, and alarm response.

A Wrong - plausible to the candidate the fails to recall that OP-AA-108-117 required protection in the field in addition to remote locations (they may assume that the MCR Operators are already aware of the importance of maintain the equipment available) and plausible to the candidate that fails to recall that Operator performing rounds, inspections and alarm response are exempted

B Wrong - plausible to the candidate the fails to recall that OP-AA-108-117 required protection in the field in addition to remote locations (they may assume that the MCR Operators are already aware of the importance of maintain the equipment available)

C Wrong - plausible to the candidate that fails to recall that Operator performing rounds, inspections and alarm response are exempted

D Correct for the above reasons

Question 69 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1845878
User-Defined ID:	Q #69
Lesson Plan Objective:	LGSOPS2010.29.A
Topic:	Recall Protected Equipment process requirements
RO Importance:	3.9
SRO Importance:	4.3
K/A Number:	2.2.14

Comments:	

General Data			
Level	RO		
Tier	3		
Group	N/A		
KA # and Rating	2.2.14 3.9 / 4.3		
KA Statement	**Knowledge of the process for controlling equipment configuration or status.**		
Cognitive level	Low		
Safety Function			
10 CFR 55	41.10		
Technical Reference with Revision No:	OP-AA-108-117	**Rev #:**	4
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

70	ID: 1846054	Points: 1.00

An Operator is performing work in the RCA. Their Electronic Dosimeter (ED) alarm set points are set at the upper limit values per RP-AA-403 (Administration of RWPs).

WHICH ONE of the following identifies the ED Alarm set points for Dose Limit and Dose Rate Limit?

	Dose Limit Alarm Setpoint	Dose Rate Limit Alarm Setpoint
A.	20 mR	80 mR/hr
B.	20 mR	400 mR/hr
C.	40 mR	80 mR/hr
D.	40 mR	400 mR/hr

Answer: D

Answer Explanation

The candidate is asked to recall the RWP set points when set to the upper limit of RP-AA-403.

From RP-AA-403, section 4.4.3:

4.4.3. Operations and Radiation Protection General RWPs:

1. Dose rate alarms should be set no higher than 400 mrem/hr.
2. Dose alarm set point should be set no higher than 40 mrem.

Distractor values are taken from step 4.4.2 of the same procedure.

4.4.2. All General RWPs **excep**t Operations and Radiation Protection:

1. Dose rate alarms. Radiation area dose rate alarms should not exceed the threshold for posting high radiation areas. Establish dose rate setpoints at 150% x highest G/A dose rate of normal walking areas for general tours and inspections not to exceed a dose rate setting of 80 mrem/hr.
2. Accumulated dose alarm: A setpoint of 125% of the anticipated dose is appropriate. If average individual doses are less than 20 mrem, set the alarm no greater than 10 mrem above the average.

A Wrong - plausible if the candidate recalls the ED setpoint for both the Dose and Dose Rate from the wrong section of RP-AA-403

B Wrong - plausible if the candidate recalls the ED setpoint for the Dose from the wrong section of RP-AA-403

C Wrong - plausible if the candidate recalls the ED setpoint for the Dose Rate from the wrong section of RP-AA-403

D Correct for the above reasons.

Question 70 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1846054
User-Defined ID:	Q #70
Lesson Plan Objective:	LLOT1760.8
Topic:	RP-AA-403 - Recall RWP ED alarm setpoints for Operators
RO Importance:	3.5
SRO Importance:	
K/A Number:	G2.3.7

Comments:

General Data			
Level	RO		
Tier	3		
Group			
KA # and Rating	G2.3.7 RO Importance 3.5		
KA Statement	G2.3.7 Ability to comply with radiation work permit requirements during normal or abnormal conditions.		
Cognitive level	Low		
Safety Function			
10 CFR 55	41.12		
Technical Reference with Revision No:	RP-AA-403	**Rev #:**	9
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Bank 986626		
Question Source: (i.e. New, Bank, Modified)	Bank 986626		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	Re-wrote stem and changed structure to 2 X 2		
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

71	ID: 1846357	Points: 1.00

A Steam Leak occurs at the Main Turbine. The following condition is present:

- Turbine Building Ventilation is shutdown

WHICH ONE of the following identifies the reason for restarting Turbine Enclosure Ventilation IAW T-104, "Radioactivity Release Control"?

A. Ensures radioactive release through elevated, monitored release point

B. Maintain negative pressure in the Turbine Enclosure

C. Provide a filtered release path to the environment

D. Ensure max safe temperature limits are NOT reached

Answer: A

Answer Explanation

Justification:

A. **Correct:** T-104 Bases states that the continued personnel access to the Turbine Enclosure and/or Radwaste Enclosure may be essential for responding to emergencies or transients which may degrade into emergencies. These areas are not always airtight structures, and a radioactivity release inside the structure would not only limit personnel access, but could eventually lead to an unmonitored ground level release. Operation of the respective HVAC system preserves accessibility and ensures radioactive discharges will be released through elevated, monitored release points.

B. **Incorrect but plausible:** Plausible, if the applicant determines that turbine building ventilation needs to be restarted to maintain negative pressure to prevent release to the environment

C. **Incorrect but plausible:** Plausible, if the applicant determines that bases behind starting the turbine building ventilation is to provide a filtered path to the environment via north stack, since turbine building ventilation exhaust to the north stack, however the basis is to provide a monitored release path not filtered release path.

D. **Incorrect but plausible:** Plausible, if the applicant determines that due to the steam leak and blowout panel actuation, temperature in the turbine building is a concern and as a result turbine building is restarted to decrease temperature

Question 71 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1846357
User-Defined ID:	Q #71
Lesson Plan Objective:	LLOT1560 OBJ. 5
Topic:	T-104 reason for re starting TE HVAC
RO Importance:	3.8
SRO Importance:	3.8
K/A Number:	G2.3.11

Comments:

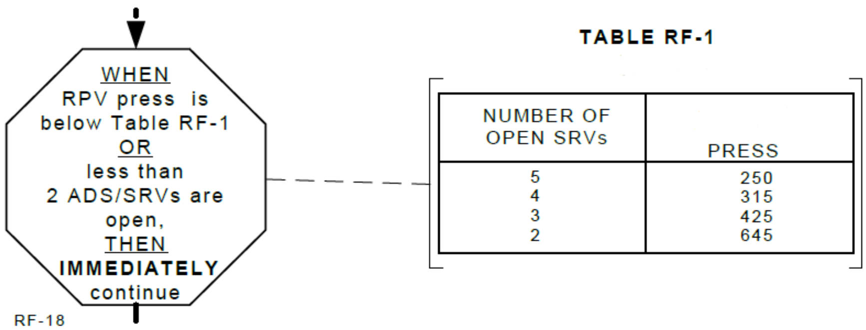
General Data			
Level	RO		
Tier	3		
Group			
KA # and Rating	2.3.11		
KA Statement	**Ability to control radiation releases**		
Cognitive level	Lower		
Safety Function			
10 CFR 55	41.11		
Technical Reference with Revision No:	T-104 Bases T-104	**Rev #:**	14 13
Justification for Non SRO CFR Link:			
Question History: (i.e. LGS NRC-05, OYS CERT-04)	2012 ILT NRC #71		
Question Source: (i.e. New, Bank, Modified)	Bank 1099858		
Low KA Justification (if required):			
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

2012 ILT NRC-71
Bank 1099858

72 | **ID: 1846393** | **Points: 1.00**

The following conditions exist for Unit 1:

RPV Level is Unknown
Reactor Power is 9%
T-116 Step RF-18 is being executed



WHICH ONE of the following identifies conditions when the Operators are directed to continue and the bases for prioritizing this steps performance by the use of "**IMMEDIATELY**"?

	Conditions requiring continuing	Bases for step priority
A.	3 SRVs Open and RPV Pressure 450 psig	adequate core cooling has been lost
B.	3 SRVs Open and RPV Pressure 450 psig	RPV is now flooded to the Main Steam Lines
C.	2 SRVs Open and RPV Pressure 575 psig	adequate core cooling has been lost
D.	2 SRVs Open and RPV Pressure 575 psig	RPV is now flooded to the Main Steam Lines

Answer: C

Answer Explanation

From the stem the candidate determines that T-116 Step RF-18 directs the performer to Immediately Continue when RPV Pressure drops below conditions of Table RF-1. (Note "MIN STM COOLING" has been covered up in the stem versions of this step)

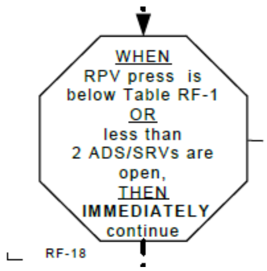


TABLE RF-1
MIN STM
COOLING PRESS

NUMBER OF OPEN SRVs	MIN STM COOLING PRESS
5	250
4	315
3	425
2	645

From the bases of T-116 RF-18:

DISCUSSION

LGS TRIP Step RF-18 is a hold/wait step and should not be exited until the condition specified in the "WHEN" statement exists.

Subsequent steps in the "ATWS" RPV flooding flowpath direct actions to flood the RPV while maintaining the core adequately cooled by a combination of submergence and steam cooling.

With RPV injection terminated and SRVs open, RPV pressure will lower. As long as pressure is above Minimum Steam Cooling Pressure (MSCP) in Table RF-1, the core will be adequately cooled by a combination of submergence and Steam Cooling regardless of whether water is being injected into the RPV or the reactor is shutdown. When RPV pressure drops below the MSCP, the core is no longer being adequately cooled and the operator must immediately continue in order to inject water into the RPV and reestablish adequate core cooling.

A Wrong - Plausible to the candiate that mis-interprets Table RF-1

B Wrong - Plausible to the candiate that mis-interprets Table RF-1 and plausible to the candidate that confuses the bases for RF-18 with the actions of non-ATWS section of T-116 step RF-35, where the operators are directed to maintain RPV injection as low as possible while maintaining the RPV flooded - this is palusible to the candiate that belives the main concern is damage to SRVs and tailpipes when injection is not reduced once the Main Steam Lines are flooded.

C Correct for the above reasons

D Wrong - Plausible to the candidate that confuses the bases for RF-18 with the actions of non-ATWS section of T-116 step RF-35, where the operators are directed to maintain RPV injection as low as possible while maintaining the RPV flooded - this is palusible to the candiate that belives the main concern is damage to SRVs and tailpipes when injection is not reduced once the Main Steam Lines are flooded.

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:	0.00
System ID:	1846393
User-Defined ID:	Q #72
Lesson Plan Objective:	LGSOPS1560.5
Topic:	Emergency Procedure Prioritization bases
RO Importance:	3.4
SRO Importance:	
K/A Number:	G2.4.23

Comments:

General Data			
Level	RO		
Tier	3		
Group			
KA # and Rating	G2.4.23 RO Importance 3.4		
KA Statement	G2.4.23 Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.		
Cognitive level	High		
Safety Function			
10 CFR 55	41.10		
Technical Reference with Revision No:	T-116 Bases	**Rev #:**	14
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

73	**ID: 1846438**	**Points: 1.00**

Initial Unit 1 plant conditions:

- Reactor power is 75%
- RPV pressure is 1009 psig

'1B' Outboard MSIV fails closed.

Current conditions:

- Reactor power is 77%
- Reactor pressure is 1045 psig

WHICH ONE of the following describes the required immediate operator action, if any?

A. No Immediate Operator Action required

B. Reduce Reactor power to ≤ 75%

C. Open BPV Jack until Reactor Pressure is ≤ 1009 psig

D. Reduce Reactor Power until Reactor Pressure is ≤ 1009 psig

Answer: B

Answer Explanation

From the stem the candidate identifies that Reactor Power and Reactor Pressure have risen as a result of the closure of the 1B MSIV. This requires entry into OT-102, Reactor High Pressure (due to Unexpected/ Unexplained rise in Reactor Pressure) and entry into OT-104, Unexpected/ Unexplained Positive or Negative Reactivity Insertion. Of these two procedures, the only Immediate Operation Action (IOA) that applies is OT-104 step 2.1 to reduce reactor power to maintain reactor power at or below the initial pre-transient level.

A Wrong - plausible to the candidate that fails to recall the OT-104 IOA to reduce power to the pre-transient level (in this case 75%) or confuses this with the followup action from OT-102 for a close MSIV to reduce power to 75% (OT-102 Attachment 1 step 1)

B Correct for the above reasons

C Wrong - plausible to the candidate the confuses the IOA of the OT-104 and OT-102 and believes that the OT-102 IOA is to reduce Reactor Pressure to or below the pre-transient pressure. (Using BPVs as described in OT-102 step 2.2)

D Wrong - plausible to the candidate the confuses the IOA of the OT-104 and OT-102 and believes that the OT-102 IOA is to reduce Reactor Pressure to or below the pre-transient pressure. (as described in OT-102 step 2.1)

Question 73 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1846438
User-Defined ID:	Q #73
Lesson Plan Objective:	LLOT1540.02
Topic:	OT-104 - Recall IOAs
RO Importance:	4.6
SRO Importance:	4.4
K/A Number:	2.4.49

Comments:	General Data			
	Level	RO		
	Tier	3		
	Group			
	KA # and Rating	G2.4.49 RO Importance 4.6		
	KA Statement	2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.		
	Cognitive level	Low		
	Safety Function			
	10 CFR 55	41.10		
	Technical Reference with Revision No:	OT-102 OT-104	**Rev #:**	2 7 5 3
	Justification for Non SRO CFR Link:	N/A		
	Question History: (i.e. LGS NRC-05, OYS CERT-04)	Modified 1006627 (LGS 2012 CERT)		
	Question Source: (i.e. New, Bank, Modified)	Modified 1006627		
	Low KA Justification (if required):	N/A		
	Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
	ILT			
	Supplied Ref (If appropriate): (i.e. ABN-##)	None		
	LORT			
	PRA: (i.e. Yes or No or #)			
	LORT Question Section: (i.e, A-Systems or B-Procedures)			
	Comments			

Question 73 Table-Item Links

NRC EXAM - 10 CFR 55 Operators' Licenses

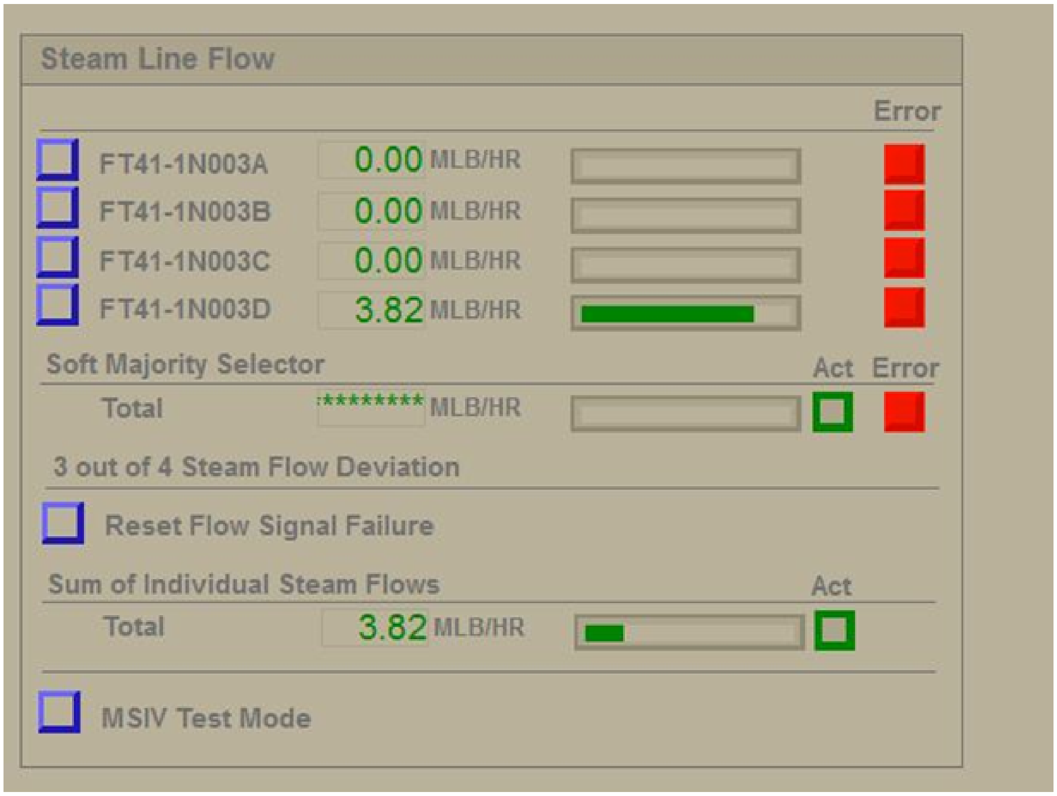
CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC EXAM - NUREG 1021 Question Cognitive Level

Memory or Fundamental Knowledge - (F)undamental

74	ID: 2029209	Points: 1.00

Unit 1 is operating at 100% power when a FWLCS trouble alarm occurs with the below indications:



Shortly thereafter, a lowering main condenser vacuum forces operators to <u>successfully</u> insert a manual scram using RPS.

WHICH ONE of the following identifies:

(1) the mode of level control (Single-element/Three-element), prior to the scram
and
(2) whether or not the Scram Profile will activate in response to the scram?

A. (1) Single-element
(2) Scram Profile will activate

B. (1) Single-element
(2) Scram Profile will NOT activate

C. (1) Three-element
(2) Scram Profile will activate

D. (1) Three-element
(2) Scram Profile will NOT activate

Answer: A

Answer Explanation	
A	Correct: The candidate should discover from the included screen shot that 3 out of 4 Steam Flow Transmitters have failed downscale. Also, a Soft Majority Select Error occurs. The candidate should recal from S06.1.H U/1, Attachment 1 Alarm List, specifically Signal Identity 1XX-FW301.ISFE. There, we find that the Steam flow Error SMS means that 3 of 4 Steam flow error signals means that FWLCS will operate in single element as steam flow in no longer available as the 3rd input. When a scram occurs and DFWLCS sees both RPS subsystems de-energize, it will force level control into post scram profile and 3 element mode. This mode will hold feedflow constant for 10 seconds and then ramp feedflow to 10% at 6% per second and then hold flow until scram profile de-activates per S06.1.D. This happens under all conditions except for a total Feedflow error (TFFE).
B	Wrong: Plausible to the candidate who recognizes that DFWLC will swap to single but believes that scram profile is not available due to the failure mechanism.
C	Wrong: Plausible to the candidate who does not recognize the conditions of total steam flow error and believes that the system remains in 3 element and scram profile will function.
D	Wrong: Plausible to the candidate who believes that the system will remain in 3 element with the single steam flow transmitter but will not support scram profile.

Question 74 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	2029209
User-Defined ID:	Q #74
Lesson Plan Objective:	LLOT0550,11
Topic:	Steam Flow Transmitter Failure - Predict Plant response
RO Importance:	3.9
SRO Importance:	3.8
K/A Number:	G 2.1.19

Comments:	General Data			
	Level	RO		
	Tier	3		
	Group			
	KA # and Rating	G2.1.19 RO Importance 3.9		
	KA Statement	G2.1.19 Ability to use plant computers to evaluate system or component status.		
	Cognitive level	High		
	Safety Function			
	10 CFR 55	41.10		
	Technical Reference with Revision No:	S06.1.H U/1 S06.1.D U/1 S06.1.D Appx 1 U/1	**Rev #:**	14 23 16
	Justification for Non SRO CFR Link:	N/A		
	Question History: (i.e. LGS NRC-05, OYS CERT-04)	Modified Bank 976742		
	Question Source: (i.e. New, Bank, Modified)	Bank		
	Low KA Justification (if required):	N/A		
	Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	Removed direct statement of Steam Flow failure and provided FWLCS screen shot requiring candidate to interpret picture		
	ILT			
	Supplied Ref (If appropriate): (i.e. ABN-##)	None		
	LORT			
	PRA: (i.e. Yes or No or #)			
	LORT Question Section: (i.e, A-Systems or B-Procedures)			
	Comments			
	#976742			

75 **ID: 1846505** **Points: 1.00**

Fuel Shuffle Part 1 is underway on Unit 2 with the following conditions:

- Spent Fuel Pools are <u>not</u> cross-connected
- Unit 2 Refuel Bridge is moving fuel from the Unit 2 core to the Unit 2 SFP
- Reactor Level is 494"

Reactor Well Seals #3 and #4 fail and current Reactor Cavity level is 475" and down slow (approximately 1"/30 minutes). All normal make-up sources have failed to inject and the CRS directs injecting with "C" LPCI.

NOTE

The following are approximate values to aid the operator in maintaining level:

Reactor Cavity	17,000 gal/ft	1,417 gal/in
Dryer/Separator Storage Pool	8,077 gal/ft	673 gal/in
Skimmer Surge Tank	400 gal/ft/unit	33 gal/in/unit
Fuel Pools (when cross-tied)	<u>8,900 gal/unit</u>	<u>742 gal/in/unit</u>
<u>Totals</u>		
Sum of above considering 1 Fuel Pool	34,377 gal/ft	2,865 gal/in
Sum of above considering Fuel Pools Cross-Connected	43,677 gal/ft	3,640 gal/in

Which ONE of the following identifies the approximate amount of time required to restore Reactor Level to 494" with rated LPCI flow?

A. 4 minutes

B. 5 minutes 30 seconds

C. 6 minutes 15 seconds

D. 8 minutes 30 seconds

Answer: B

Answer Explanation

From the stem the candidate determines there is a leak from the reactor cavity and ON-120 should be implemented. Also from the stem the candidate determines that Cavity Level needs to be raised 19". LPCI rated injection rate is 10,000 GPM.

Per the attached note from ON-120, with the fuel pools not cross-connected, the volume per inch is 2865 gal/inch. Since level has dropped 19", the volume to make up is 19 x 2865 = 54,435 gallons. LPCI is rated for 10,000 gpm, so operation for approx. 5 minutes and 30 seconds would restore reactor level.

A Wrong - plausible to the candidate that incorrectly uses the volume of the reactor cavity and dryer/separator storage pool only

B Correct for the above reasons

C Wrong - plausible to the candidate that incorrectly uses the volume of the reactor cavity and dryer/separator storage pool only and uses the rated flow rate for a loop of Core Spray (6,350 GPM)

D Wrong - plausible to the candidate that incorrectly uses the rated flow rate for a loop of Core Spray (6,350 GPM)

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:	0.00
System ID:	1846505
User-Defined ID:	Q #75
Lesson Plan Objective:	LGSOPS1550.3
Topic:	Refueling - lowering cavity level
RO Importance:	3.8
SRO Importance:	
K/A Number:	G2.4.9

Comments:

General Data			
Level	RO		
Tier	3		
Group			
KA # and Rating	G2.4.9 RO Importance		
KA Statement	G2.4.9 Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.		
Cognitive level	High		
Safety Function			
10 CFR 55	41.10		
Technical Reference with Revision No:	ON-120	**Rev #:**	28
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Bank 833372		
Question Source: (i.e. New, Bank, Modified)	Bank 833372		
Low KA Justification (if required):			
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)	Added rate of level lowering (1"/30min) to inform the candidate that rate of level lowering is negligible for this question. Added table from ON-120.		

ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	None

LORT	
PRA: (i.e. Yes or No or #)	.
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

76	ID: 1743064	Points: 1.00

**********SRO ONLY**********

Unit 2 is in OPCON 4 with the following conditions:

- The reactor has been shutdown for 20 days
- "2A" RHR pump is operating in Shutdown Cooling Mode
- "2B" RHR pump is inoperable and unavailable
- RPV water temperature is currently 120° F

At time 14:00, a loss of 2A-Y160 occurs

At time 14:20 RPV coolant temperature is 130° F and rising slowly

WHICH ONE of the following describes the required operator action, and the estimated time when Rx Enclosure Secondary Containment is required?

	Required Action	Time when RE Sec. Containment Required per TS LCO 3.6.5.1.1
A.	Perform attachment 1 of ON-121 for loss of 2A-Y160	16:40
B.	Perform attachment 1 of ON-121 for loss of 2A-Y160	17:00
C.	Place "2C" RHR in Shutdown Cooling per S51.8.H, Use of Dedicated LPCI Pumps for Shutdown Cooling/Reactor Coolant Circulation Operation	16:40
D.	Place "2C" RHR in Shutdown Cooling per S51.8.H, Use of Dedicated LPCI Pumps for Shutdown Cooling/Reactor Coolant Circulation Operation	17:00

Answer: A

Answer Explanation

Source for 212 °F distractor:

3/4.10.8 INSERVICE LEAK AND HYDROSTATIC TESTING

LIMITING CONDITIONS FOR OPERATION

3.10.8 When conducting inservice leak or hydrostatic testing, the average reactor coolant temperature specified in Table 1.2 for OPERATIONAL CONDITION 4 may be increased to 212°F, and operation considered not to be in OPERATIONAL CONDITION 3, to allow performance of an inservice leak or hydrostatic test provided the following OPERATIONAL CONDITION 3 Specifications are met:

a. 3.3.2 ISOLATION ACTUATION INSTRUMENTATION, Functions 7.a, 7.c.1, 7.c.2 and 7.d of Table 3.3.2-1;
b. 3.6.5.1.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY;
c. 3.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY;
d. 3.6.5.2.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES;
e. 3.6.5.2.2 REFUELING AREA SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES; and
f. 3.6.5.3 STANDBY GAS TREATMENT SYSTEM.

APPLICABILITY: OPERATIONAL CONDITION 4, with average reactor coolant temperature greater than 200°F and less than or equal to 212°F.

a Correct: ON-121 directs the performance of Attachment 1 for The Loss of 2AY160. This will jumper power to the valve open contactors of the Inboard isolation valve (009) and allow reset of the isolation to the outboard valve. This will allow restoration of SDC. With the loss of cooling and the provided temperature changes, the candidate should calculate the heatup rate as 30°F per hour. With a starting point of 120°F, this means 200°F will be reached in 2.66667 hours or 2 hours 40 minutes (time 16:40). At 200°F, a mode change occurs from mode 4 to mode 3 and secondary containment is required to be operable per Tech spec. 3.6.5.1.1. ON-121 step 2.1.3 also reminds the operator that secondary containment is required prior to changing modes.

b Wrong: Plausible to the candidate who calculates the heat up rate to 212° (Tech Spec 3.10.8 instead of 200°F.

c Wrong: Plausible to the candidate who fails to consider that the loss of 2AY160 would not allow the use of the same suction flowpath.

d Wrong: Plausible to the candidate who fails to consider that the loss of 2AY160 would not allow the use of the same suction flowpath and who calculates the time to 212°F.

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:	0.00
System ID:	1743064
User-Defined ID:	Q #76
Lesson Plan Objective:	LGSOPS1550.02 (ON-121)
Topic:	(SRO Only) Loss of SDC and Time to Boil
RO Importance:	3.6
SRO Importance:	3.6
K/A Number:	295021 AA2.01

Comments:	General Data			
	Level	SRO		
	Tier	1		
	Group	1		
	KA # and Rating	295021 AA2.01 3.5 / 3.6		
	KA Statement	**Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING:** Reactor water heatup/cooldown rate		
	Cognitive level	Hi		
	Safety Function	4		
	10 CFR 55	CFR: 43.5 (b)(5)		
	Technical Reference with Revision No:	ON-121	**Rev #:**	32
	Justification for Non SRO CFR Link:			
	Question History: (i.e. LGS NRC-05, OYS CERT-04)			
	Question Source: (i.e. New, Bank, Modified)	New		
	Low KA Justification (if required):			
	Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
	ILT			
	Supplied Ref (If appropriate): (i.e. ABN-##)	None		
	LORT			
	PRA: (i.e. Yes or No or #)			
	LORT Question Section: (i.e, A-Systems or B-Procedures)			
	Comments			

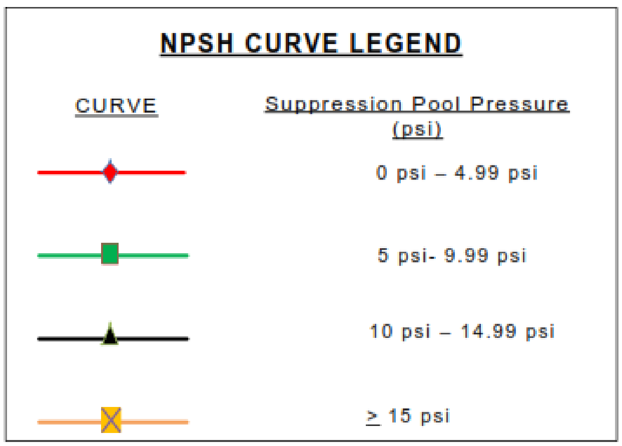
77	ID: 1744158	Points: 1.00

**********SRO ONLY**********

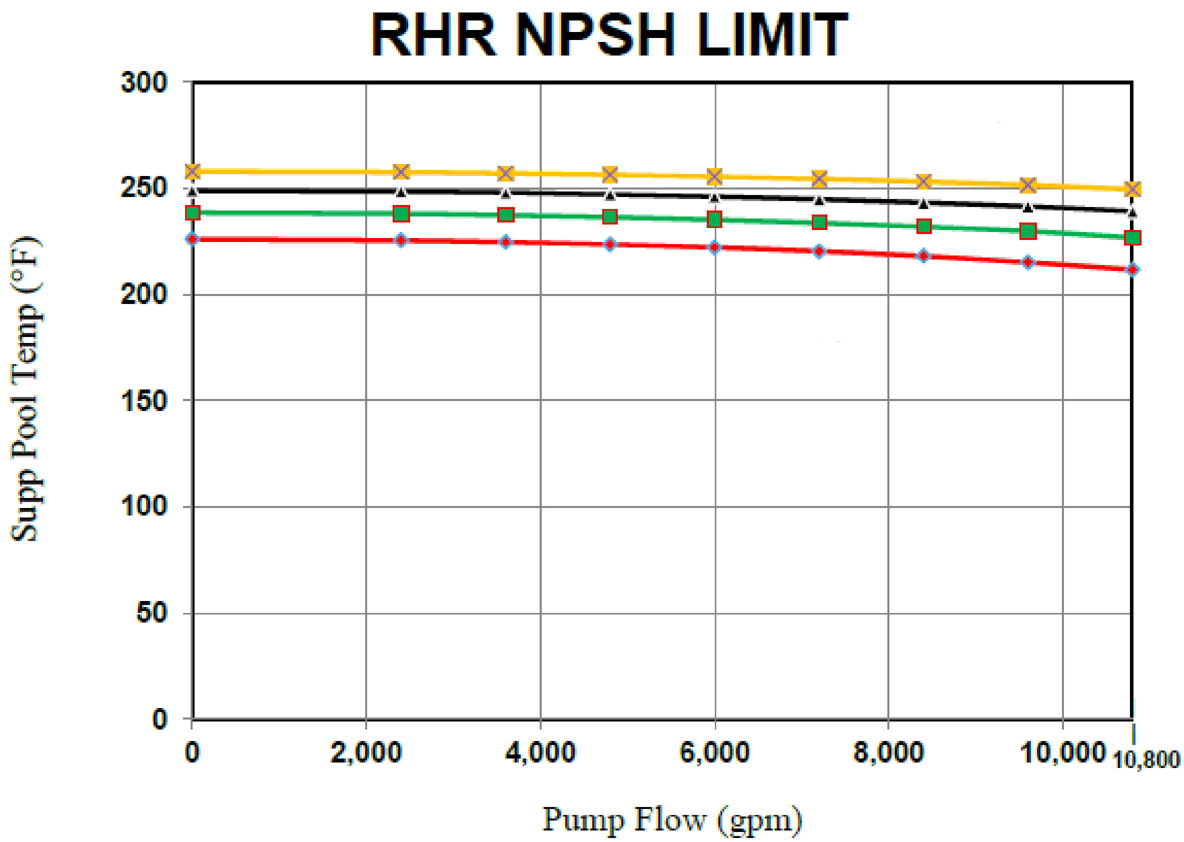
Plant conditions are as follows:

- Suppression Pool temperature is 255° F
- Suppression Pool pressure is 14.8 psig up slow
- Suppression Pool level is 31.5 feet

Given the above conditions: What actions are required per T-225, "Startup and Shutdown of Drywell and Suppression Pool Spray", regarding Drywell Spray?



-



A. Spray the Drywell with RHR with No Limitations

B. Spray the Drywell with RHR Flow <9000 GPM to prevent exceeding NPSH Limits

C. Spray the Drywell with RHRSW because RHR NPSH Limits are exceeded

D. Don't Spray the Drywell, Suppression Pool Level is too high

Answer: C

Answer Explanation

A Incorrect, plausible to the candidate who reads the RHR pump NPSH curve and interprets the area above the line as safe for pump operation and sprays.

B Incorrect plausible to the candidate who uses the top curve for S.P. pressure and determines that flow below 9000 GPM is acceptable.

C Correct, Spray the drywell with RHRSW as RHR is unavailable because SP Temperature is unsafe for NPSH. RHRSW should be used since level in Supp. Pool is below the high SP level requiring securing of drywell sprays. (37.4')

D Incorrect, Plausible to the candidate who reads the NPSH curve correctly but fails to recall the level at which Drywell Spray should be secured. Per T-102, SP/L-17, when SP water level > 37.4 ft, then SP spray should be secured. Above that level, the vacuum breakers would be obstructed and prevent pressure equalization between drywell and suppression pool.

Question 77 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1744158
User-Defined ID:	Q #77 NEW
Lesson Plan Objective:	LGSOPS0051.20
Topic:	SRO Only - Suppression Pool Temperature
RO Importance:	4.1
SRO Importance:	4.1
K/A Number:	295024 EA2.06

Comments:	

General Data			
Level	SRO		
Tier	1		
Group	1		
KA # and Rating	295024 EA2.06 4.1 / 4.1		
KA Statement	**Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE:** Suppression pool temperature		
Cognitive level	higher		
Safety Function	5		
10 CFR 55	43.5(b)(5)		
Technical Reference with Revision No:	T-102	**Rev #:**	25
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	N/A		
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

78 **ID: 1744190** **Points: 1.00**

***********SRO ONLY************

Plant conditions:

- Both Units are operating at 100% power
- 10 Station Aux Bus Feeder Breaker, 105, tripped 4 hours ago and the associated offsite source was declared INOPERABLE
- All 8 Safeguard Buses are aligned to the 201 Bus

The TSO notifies LGS that the 500 KV Substation voltage is 497 KV.

WHICH ONE of the following identifies the Tech Spec required action?

A. Restore one offsite circuit to OPERABLE within 24 hours and the second offsite circuit to OPERABLE within 68 hours.

B. Restore the 10 Bus Source to OPERABLE within 68 hours and the 20 Bus Source to OPERABLE within 72 hours.

C. Restore one AC source to OPERABLE within 8 hours and the second offsite circuit to OPERABLE within 72 hours.

D. Within one hour take action to place both units in STARTUP within the next 6 hours, HOT SHUTDOWN within the following 6 hours and COLD SHUTDOWN within the subsequent 24 hours.

Answer: A

Answer Explanation

The voltage provided in the stem would be the result of Grid Disturbance.

A Correct: With one (1) offsite circuit INOPERABLE with either unit in OPCONS 1, 2 OR 3, the INOPERABLE offsite circuit must be restored to OPERABLE within 72 hours per Tech Spec 3.8.1, action "f'. With the MAX GRID EMERGENCY GENERATION action entered and 500 KV Substation less than 498 KV, the 20 Bus Source must be declared INOPERABLE per E-5, Grid Emergency. With two (2) offsite circuits INOPERABLE, Tech Spec 3.8.1, action "g" applies. Action "g" requires restoration of at least one (1) offsite circuit within 24 hours and two (2) offsite circuits restored to OPERABLE within 72 hours from the initial loss. Since the first loss occurred 4 hours previous, per the tech spec action, the second source must be restored within 68 hours (72-4=68).

B Wrong: Plausible to the candidate who misreads 3.8.1.f and believes that each offsite source can be INOP for 72 hours

C Wrong: Plausible to the candidate who believes 1 offsite source and 2 EDGs are INOP and this would make action 3.8.1.h. plausible. The candidate could determine that the 20 bus was not INOP but the loss of the 10 bus, which resulted in auto start of the and start of the EDGs INOPed 1 offsite source and 2 EDGs.

D Wrong: Plausible to the candidate who believes that both offsite sources and 2 EDGs are INOP. No action exists in section 3.8.1 for this condition and therefore the actions of 3.0.3 would apply.

Question 78 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1744190
User-Defined ID:	Q #78
Lesson Plan Objective:	LGSOPS0036A.2
Topic:	(SRO Only) - Both Offsite Sources INOP
RO Importance:	3.2
SRO Importance:	3.8
K/A Number:	700000 AA2.05

Comments:	

General Data			
Level	SRO		
Tier	1		
Group	1		
KA # and Rating	700000 AA2.05		
KA Statement	**Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES:** Operational status of offsite circuit		
Cognitive level	Higher		
Safety Function	6		
10 CFR 55	CFR: 43.5 (b)(2)		
Technical Reference with Revision No:	E-5	**Rev #:**	23
Justification for Non SRO CFR Link:			
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Bank 633775		
Question Source: (i.e. New, Bank, Modified)	Bank		
Low KA Justification (if required):			
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)			
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

79	ID: 1744191	Points: 1.00

**********SRO ONLY**********

Both Unit 1 and Unit 2 are at 100% power with the following equipment out of service:

- 0D ESW Pump - was declared INOP 2 days ago

D12 EDG was started for ST-6-092-312-1, D12 Diesel Generator Slow Start Operability Test Run.

- The PRO identifed that the associated ESW Pump was not running 1 minute and 20 seconds after the D12 EDG start. The PRO successfully started the associated ESW Pump manually using the MCR handswitch.

For the above conditions, what is the most limiting Tech Spec Action?

A. Restore the inoperable pump to OPERABLE status within 43 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

B. Restore at least one inoperable pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

C. Restore the inoperable loop to OPERABLE status with at least one OPERABLE pump within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

D. Within one hour initiate action to place the unit in an OPERATIONAL CONDITION in which the Specification does not apply by placing it in at least STARTUP within the next 6 hours,at least HOT SHUTDOWN within the following 6 hours, and at least COLD SHUTDOWN within the subsequent 24 hours.

Answer: C

Answer Explanation

A Wrong, plausible to the candidate that fails to declare the 0B ESW Pump INOP (believes that since is was successfully started it remains Operable) and applies a continuation of Tech Spec 3.7.1.2 action a.1.

B Wrong, plausible to the candidate that confuses the associated ESW loop being impacted by the loss of the 0B ESW pump and believes that the 0D is in the other loop ESW cooling loop.

C Correct: From the stem, 0D ESW Pump is INOP. TS 3.7.1.2 action a applies. Also from the stem, during the D12 EDG start the associated ESW Pump failed to auto start (should have started at ~55 seconds, during the time between the auto start failure and the manual start of the 0B ESW Pump, the B ESW Loop experienced a complete loss of Component Cooling Water (CCW)). For D12 the candidate determines that the associated ESW Pump is the 0B ESW Pump.

Tech Spec surveillance requirement 4.7.1.2.b.2 requires ESW pumps to start automatically when its associated diesel generator starts.

D Wrong, plausible to the candidate that fails to determine that Tech Spec action 3.7.1.2.a.3 applies to 2 ESW pumps INOP in the same loop and incorrectly applies Tech Spec 3.0.3.

Since the 0B ESW Pump failed to automatically start, Tech spec 4.0.1 requires the 0B ESW Pump to be declared INOP.

0B and 0D ESW Pumps both support the B ESW Loop. With both ESW Pumps in the 0B ESW Loop INOP, Tech Spec Action 3.7.1.2.a.3 applies.

Question 79 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1744191
User-Defined ID:	Q #79
Lesson Plan Objective:	LLOT1870.01I
Topic:	SRO Only degraded ESW and affect on Operability
RO Importance:	3.6
SRO Importance:	4.6
K/A Number:	G.2.2.37

Comments:	General Data			
	Level	SRO		
	Tier	2		
	Group	1		
	KA # and Rating	295018 G.2.2.37 SRO 4.6		
	KA Statement	295018 Partial or Complete Loss of Component Cooling Water G.2.2.37 Ability to determine operability and/or availability of safety related equipment.		
	Cognitive level	High		
	Safety Function	8- Plant Service Systems		
	10 CFR 55	43 (b)(2)		
	Technical Reference with Revision No:	T.S.3.7.1.2	**Rev #:**	
	Justification for Non SRO CFR Link:	N/A		
	Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
	Question Source: (i.e. New, Bank, Modified)	New		
	Low KA Justification (if required):	N/A		
	Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
	ILT			
	Supplied Ref (If appropriate): (i.e. ABN-##)	Tech Spec 3.7.1.2 (Pages 3/4 7-3 and 7-4)		
	LORT			
	PRA: (i.e. Yes or No or #)			
	LORT Question Section: (i.e, A-Systems or B-Procedures)			
	Comments			

80	ID: 1744192	Points: 1.00

**********SRO ONLY*********

Both Units are at 100% power when the following occurs:

- A Loss of All Offsite Power
- All Unit 1 Diesel Generators have failed to start
- Unit 1 Reactor level is -40" and rising
- Unit 1 HPCI and RCIC are running and injecting

Given:

- S55.1.D, HPCI System Full Flow Functional Test
- S55.2.A, HPCI Shutdown From Automatic or Manual Initiation

WHICH ONE of the following describes 1) the instrument that can be used to determine Unit 1 Reactor pressure per E-1, Station Blackout and 2) what time critical action the crew must take?

	Instrument Used	Time Critical Action
A.	"A" PAMS, XR-42-1R623A	Swap HPCI suction to the Suppression Pool per S55.1.D
B.	"A" PAMS, XR-42-1R623A	Shutdown HPCI per S55.2.A
C.	RCIC Steam Pressure, PI-49-1R602	Swap HPCI suction to the Suppression Pool per S55.1.D
D.	RCIC Steam Pressure, PI-49-1R602	Shutdown HPCI per S55.2.A

Answer: D

Answer Explanation

The Station Blackout is described in section 15.12 of the LGS USFAR Chapter 15 - Accident Analysis.

Station blackout is addressed by Limerick procedure E-1, Loss of all AC Power (Station Blackout)

During a station blackout very few instruments remain available for monitoring the reactor during the accident. They are identified by E-1.

From E-1 BASES

3.3 **MONITOR** plant parameters using available instrumentation listed in Attachment 1. ________

3.3.1 **IF** critical DC powered instruments are not available, **THEN** implement appropriate section of T-370 to establish alternate monitoring methods. ________

BASIS

Attachment 1 lists the key instrumentation supplied by DC power, which will remain availa during a station blackout or ELAP.

Instrumentation in Attachment 1 related to RPV parameters, primary containment, a operation of the RCIC system should be closely monitored in order to cope with a stati blackout and subsequent reactor scram.

APRMs are utilized upon entry into T-101, RPV Control, or T-100, Scram/ Scram Recove in order to confirm that reactor scram has successfully shutdown the reactor.

RPV level instrumentation is used to ensure that the RCIC system is adequately maint RPV water level in the required operating band of +12.5 to +54 inches specifically in st 3.2.3 and is consistent with the desired RPV water level band specified in T-100 and T-10

Primary Containment parameters related to drywell pressure, suppression pool level a suppression pool temperature are closely monitored since the design basis station black will require entry into T-102, "Primary Containment Control". Although equipment requir to respond to elevated primary containment parameters may not be readily available in t early steps of a station blackout, when equipment is restored then operator actions per 102 may be implemented.

The FLEX electrical strategies will repower the Division 1 and 2 battery chargers to ensu the critical instruments can remain functional indefinitely. Procedure T-370 has be developed to document alternate methods to obtain critical instrument parameters durin loss of AC power event combined with a DC instrument failure. These strategies re transmitters directly using a self-contained calibrator or read RTDs using a multimeter. T Shift I&C Techs may be used to connect the instruments used for alternate instrum strategies.

The following RPV pressure instruments are available during a station blackout:
PI-42-*R605 WR
PI-55-*R602 HPCI (available)
PI-49 *R602 RCIC (available)

A. Incorrect but plausible to the candidate who remembers that "A" PAMS is a EQ Post Accident indicator but forgets that it loses power on a Station Blackout.

Question 80 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1744192
User-Defined ID:	Q #80
Lesson Plan Objective:	LGSOPS2000.02
Topic:	(SRO Only) Determine RPV pressure instr. available during a Station Blackout and Required Actions
RO Importance:	
SRO Importance:	3.9
K/A Number:	295025 G.2.4.3

Comments:

General Data			
Level	SRO		
Tier	1		
Group	1		
KA # and Rating	295025 G.2.4.3		
KA Statement	**High Reactor Pressure:** Ability to identify post-accident instrumentation		
Cognitive level	Lower		
Safety Function	3		
10 CFR 55	43(b)(5)		
Technical Reference with Revision No:	SE-1	**Rev #:**	50
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

81	ID: 1744196	Points: 1.00

**********SRO ONLY*********

Unit 2 was at 100% power when a FWLCS failure occurred.

- During the failure RPV level was as low as -60".
- The Reactor Mode Switch was placed in Shutdown when RPV Level was -20"
- RPV level has been recovered to 30" with HPCI, RCIC, and manual control of Feedwater
- Reactor Power is steady at 45%

In this event, 1) What is the basis for the RPS Setpoint that was exceeded first and 2) when is HPCI required to be secured?

	Setpoint Basis	When HPCI required to be secured
A.	Assure that there is adequate protection for the fuel	HPCI is required to be secured immediately
B.	Assure that there is adequate protection for the fuel	HPCI is permitted to continue injecting until T-270 is directed
C.	Reduce the amount of energy being added to the coolant	HPCI is required to be secured immediately
D.	Reduce the amount of energy being added to the coolant	HPCI is permitted to continue injecting until T-270 is directed

Answer: B

Answer Explanation

From the stem the candidate determines that the RPV Low Level RPS setpoint (12.5") was exceeded and the reactor should have shutdown. Also from the stem the candidate determines that the scram was unsuccessful.

T-101 initial entry was RPV Level below 12.5". The basis for the 12.5" RPS Setpoint is found in Tech Spec 2.2.1 Bases Page B 2-8.

BASES

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint has been used in transient analyses dealing with coolant inventory decrease. The scram setting was chose far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

From the stem the candidate is given that RCIC and HPCI have started and are injecting (initiation based on RPV Level below -38"). The candidate now needs to recall the basis of T-117 step LQ-7 to conclude that HPCI is permitted to inject until T-270 is directed.

DISCUSSION

LGS TRIP Step LQ-7 is a continue re-checking step, and as such, sho be referred to frequently to determine if both of the conditions listed exist, and if so, to carry out the specified action.

Step LQ-7 directs actions to deliberately lower RPV level below -50 inches when reactor power is above 4% or cannot be determined.

The HPCI system normally injects through the Feedwater (outside the core shroud) and Core Spray (inside the Core shroud) injection line Running HPCI at rated flow during an ATWS may be necessary to preve excessive reactor level lowering, closure of the MSIVs, and uncover of fuel. If not required for reactor level control, HPCI should be minimized to limit core inlet sub-cooling and reduce reactor power. HPCI system injection through the "B" core spray line in an ATWS wa evaluated during initial licensing and construction of the facility UFSAR section 15.8.3.7 documents the fact that the HPCI system was analyzed to inject during an ATWS without challenging the integrity the fuel and references GE analysis NEDE-24222. In addition, the AT analysis completed for the MUR power uprate project (LEAM-MUR-0055) evaluated injection from HPCI through the "B" core spray line durin an ATWS.

A Wrong - Plausible to the candidate that fails to recall that HPCI injection during an ATWS has been analyzed to continue due to split flow path
B Correct for the above reason
C Wrong - plausible to the candidate that recalls the basis for the RPS Setpoint for Drywell Pressure - High rather than the correct basis for RPV Low Level and plausible to the candidate that fails to recall that HPCI injection during an ATWS has been analyzed and is permitted until trips direct secure.
D Wrong - plausible to the candidate that recalls the basis for the RPS Setpoint for Drywell Pressure - High rather than the correct basis for RPV Low Level

Question 81 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1744196
User-Defined ID:	Q #81
Lesson Plan Objective:	LLOT1560 #6
Topic:	SRO Only - ATWS Basis for Entry Conditions and Automatic Actions
RO Importance:	
SRO Importance:	4.6
K/A Number:	295037 2.4.2

Comments:

General Data			
Level	SRO		
Tier	1		
Group	1		
KA # and Rating	295037 2.4.2 SRO Importance 4.6		
KA Statement	295037 SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown / 1 2.4.2 - Emergency Procedures / Plan: Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.		
Cognitive level	High		
Safety Function	1 - Reactivity Control		
10 CFR 55	43(b)(2)		
Technical Reference with Revision No:	Tech Spec 2.2.1 T-117 Bases	**Rev #:**	0 1 9
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):			
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

82	ID: 2027351	Points: 1.00

***********SRO ONLY***********

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

- The Unit 1 RO observes rising RPV Pressure
- The Reactor Mode Switch is placed in Shutdown when RPV Pressure is 1085 psig
- A group 1 MSIV Isolation occurs
- RPS fails to de-energize
- After 35 seconds the RO reports all rods in and also reports that peak RPV Pressure was 1317 psig
- No injection sources are available

2 hours later the CRS is implementing the Steam Cooling section of T-111.

WHICH ONE of the following identifies (1) the safety limit violated, and (2) correctly completes the below statement from Tech Spec 6.7.1.a?

6.7.1.a The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within ___(2)___.

	Safety Limit Violated	(2)
A.	Reactor Coolant System Pressure	15 minutes
B.	Reactor Coolant System Pressure	1 hour
C.	Reactor Vessel Water Level	15 minutes
D.	Reactor Vessel Water Level	1 hour

Answer: D

Answer Explanation

From the stem the candidate determines that during the transient Unit 1 entered OPCON 3 when the mode switch was placed in Shutdown and that RPV pressure peak at 1317 psig. This is below the Safety Limit value of 1325 psig. Also from the stem the candidate concludes that RPV level is below -161 inches (Top of Active Fuel) and above is -198 inches (the RPV level where the steam cooling section of T-111 is exited (Step LR-17). Based on this information the Reactor Vessel Water Level Safety Limit has been exceeded.

Excerpts from Tech Spec sections 2 and 6

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vesse steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with the reactor co system pressure less than or equal to 1325 psig within 2 hours and comply wi the requirements of Specification 6.7.1.

REACTOR VESSEL WATER LEVEL

2.1.4 The reactor vessel water level shall be above the top of t active irradiated fuel.

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4, and 5.

ACTION:

With the reactor vessel water level at or below the top of the ac irradiated fuel, manually initiate the ECCS to restore the water after depressurizing the reactor vessel, if required. Comply wit requirements of Specification 6.7.1.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit violated:

a. The NRC Operations Center shall be notified by telephone as soon possible and in all cases within 1 hour. The Vice President, Lim Generating Station, Plant Manager, and the NRB shall be notified 24 hours.

A Wrong - plausible to the candidate that fails to recall the safety limit set points and/or the correct applicability for the associated Safety Limit and plausible to the candidate the confuses the time limit for notifying the state and local agencies (15 minutes) when declaring an emergency from the Emergency Plan with the time requirement for notifying the NRC.

B Wrong - plausible to the candidate fails to recall the safety limit set points and/or the correct applicability

C Wrong - plausible to the candidate the confuses the time limit for notifying the state and

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:	0.00
System ID:	2027351
User-Defined ID:	Q #82
Lesson Plan Objective:	LGSOPS1800.07C
Topic:	SRO ONLY - Determine Safety Limit Violated and Reporting action Time
RO Importance:	
SRO Importance:	4.7
K/A Number:	295031 2.2.22

Comments:	

General Data			
Level	SRO		
Tier	1		
Group	1		
KA # and Rating	295031 2.2.22 SRO Importance 4.7		
KA Statement	295031 Reactor Low Water Level 2.2.22 - Knowledge of limiting conditions for operations and safety limits.		
Cognitive level	High		
Safety Function	2 - Reactor Water Inventory Control		
10 CFR 55	43(b)(1)		
Technical Reference with Revision No:	Tech Specs sections 2.0 and 6.0 T-111	**Rev #:**	1 7
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):			
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

83	ID: 2027489	Points: 1.00

********************SRO ONLY********************

Unit 2 plant conditions are as follows:

- GP-2, Normal Plant Startup, in progress
- RPV pressure is 850 psig
- "2A" Control Rod Drive (CRD) pump is out for maintenance

"2B" CRD pump trips, and HCU 30-11, CRD accumulator trouble alarms simultaneously

3 minutes later, HCU 18-27, CRD accumulator trouble alarms

Both control rods 30-11 and 18-27 are verified at position 08, and accumulator pressures are:

HCU 30-11 - 925 psig
HCU 18-27 - 960 psig

WHICH ONE of the following describes the minimum REQUIRED action(s) for the above condition?

A. Restore the HCU accumulator 30-11 to Operable within 8 hrs

B. Place the mode switch in SHUTDOWN

C. Manually scram the reactor and place the mode switch in SHUTDOWN if a CRD pump cannot be started within 20 minutes

D. Manually scram the reactor and place the mode switch in SHUTDOWN if a third accumulator trouble alarm comes in prior to starting a CRD pump

Answer: A

Answer Explanation

A Correct: Accumulator 30-11 is INOP because pressure is below 955 psig
Accumulator 18-27 is OPERABLE (alarm set point is 970 psig)
Per ON-107 attachment 5 step 1: **IF** one CRD scram accumulator is inoperable,
THEN PERFORM the following within 8 hours:
1.1 RESTORE the inoperable accumulator to operable
OR DECLARE associated control rod inoperable.

B Wrong: Plausible if the student identifies the action for reactor pressure less than 900 psig for 2 accumulators INOP

C Wrong: Plausible if the student identifies the action for reactor pressure greater than 900 psig for 2 accumulators INOP

D Wrong: Pausible if the student misses that the accumulator trouble alarm set point is 970 and does not necessarily mean the pressure is below the tech spec value.

Question 83 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	0.00
System ID:	2027489
User-Defined ID:	Q #83
Lesson Plan Objective:	LLOT1550.3
Topic:	(SRO) GP-2, Normal Plant Startup, in progress - Reactor pressure is 850 psig - "2A" CRD PP
RO Importance:	
SRO Importance:	3.6
K/A Number:	295022 AA2.01

Comments:

General Data			
Level	SRO		
Tier	1		
Group	2		
KA # and Rating	295022 AA2.01 SRO Importance 3.6		
KA Statement	295022 Loss of CRD Pumps AA2.01 - Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS : Accumulator pressure		
Cognitive level	High		
Safety Function	1 - Reactivity Control		
10 CFR 55	43(b)(5)		
Technical Reference with Revision No:	ON-107	**Rev #:**	19
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Modified 558686		
Question Source: (i.e. New, Bank, Modified)	Modified 558686		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

84 **ID: 1744249** **Points: 1.00**

**********SRO ONLY*********

Unit 1 conditions are as follows:

- Unit 1 was scrammed due to an unisolable leak into the RWCU Room
- Division 1 DC power has been lost
- Both EHC Pumps have tripped

The EO performing T-290 reports the following temperatures:

Area	TEMP(°F)
Room 309	127
RWCU Regen htx room	123
A nonregen htx room	118
B nonregen htx room	124
A RWCU pump room	168
B RWCU pump room	150
C RWCU pump room	149
Room 510	149

TABLE SCC-2

Max Safe Op (MSO) Values

AREA	MAX SAFE OP VALUE			
	TEMP(°F)	RAD(mr/hr)	WATER LEVEL (IN.)	
			UNIT 1	UNIT 2
HPCI	176	10,000	15	29
RCIC	155	10,000	27	40
A & C RHR	140	10,000	18	18
B & D RHR	140	10,000	18	18
A & C Core Spray	140	----------------	(A) 12 (C) 19	(A) 12 (C) 17
B & D Core Spray	140	----------------	(B) 12 (D) 19	(B) 12 (D) 19
HPCI/RCIC Pipeway Rm 309/376	145	10,000	17	30
Safeguard sys Access Rm 304/370	----------------	10,000	11	11
RWCU regen htx room	120	10,000	----------------	
A & B nonregen htx room	120	10,000	----------------	
A,B,& C RWCU pump room	158	10,000	----------------	
Isolation valve compartment Rm 510 & 522/ 584 & 597	135	----------------	----------------	
Outboard MSIV room	145	10,000	----------------	

Given:

- T-101, RPV Control
- T-112, Emergency Blowdown

Given the above conditions, WHICH ONE of the following identifies the correct pressure control strategy required by TRIPs, and the location where the RO can achieve this strategy?

	Pressure Control Strategy Required by TRIPs	Remote SRV Operation Location
A.	Depressurize RPV within 100°F/hr per T-101	Auxiliary Equipment Room
B.	Depressurize RPV within 100°F/hr per T-101	Remote Shutdown Panel
C.	Perform Emergency Blowdown per T-112	Auxiliary Equipment Room
D.	Perform Emergency Blowdown per T-112	Remote Shutdown Panel

Answer: A

Answer Explanation

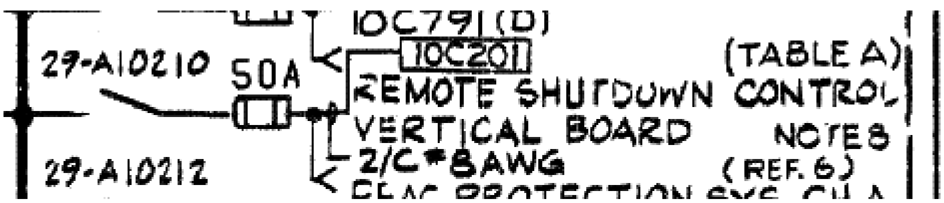
From the stem the candidate determines that T-103, Secondary Containment Control, has been entered and the RWCU area ONLY is above MSO. Also from the stem DIV 1 DC has no power. From E-1FA, DIV 1 ADS AND all non-ADS SRVs cannot be activated.

NOTE

1. INBD MSIV Indication will be lost. [
2. RCIC **cannot** be initiated. [
3. DIV 1 ADS
AND all non- ADS SRV(s) **cannot** be activated. [

With EHC pumps off, bypass valves are not available. Pressure control is available using ADS SRVs only and only from the DIV 3 controls in the Auxilary Equipment Room. The SRV controls for the C,A,and N SRVs at the Remote Shutdown Panel are Division 1.

From E-0033 Sheet 1:



From T-101, step RC/P-22, Maintain cooldown rate less than 100°F/hr.

A Correct for the above reasons

B Wrong - plausible to the candidate that incorrectly recalls the power supplying SRV control to the Remote Shutdown Panel as Division 3.

C Wrong - plausible to the candidate that incorrectly recalls T-103 step SCC/T-10 and believes that an Emergency Blowdown is required when 2 or more temperatures exceed MSO as opposed to the correct requirement of 2 or more Areas

D Wrong - plausible to the candidate that incorrectly recalls T-103 step SCC/T-10 and believes that an Emergency Blowdown is required when 2 or more temperatures exceed MSO as opposed to the correct requirement of 2 or more Areas, and Wrong - plausible to the candidate that incorrectly recalls the power supplying SRV control to the Remote Shutdown Panel as Division 3.

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:	0.00
System ID:	1744249
User-Defined ID:	Q #84
Lesson Plan Objective:	LGSOPS1560.02D
Topic:	SRO Only - Secondary Containment high temp - RO actions outside the MCR
RO Importance:	
SRO Importance:	4.1
K/A Number:	295032 - 2.4.34

Comments:

General Data			
Level	SRO		
Tier	1		
Group	2		
KA # and Rating	295032 - 2.4.34 - SRO 4.1		
KA Statement	295032 High Secondary Containment Area Temperature 2.4.34 - Emergency Procedures / Plan: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.		
Cognitive level	High		
Safety Function	5 - Containment Integrity		
10 CFR 55	43 (b) 5		
Technical Reference with Revision No:	T-103 E-1FA E-0033 Sheet 1	**Rev #:**	1 5
Justification for Non SRO CFR Link:			
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):			
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

85	ID: 1740796	Points: 1.00

**********SRO ONLY*********

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

- Eight Drywell cooling fans are running.
- 1A Drywell Chiller is operating
- Drywell temperature is 125°F and steady
- Drywell pressure is 0.5 psig and steady.

An inadvertent Group 8A isolation occurs that cannot be reset.

- Drywell pressure is 0.7 psig up slow,
- Drywell temperature is 130°F up slow
- G532, DWCW Head Tank Level indicates Hi/Lo Level
- Pressure on PI-087-109A, "CHW CIRCULATION PUMP 1AP161", is 5 psig

WHICH ONE of the following describes the Required Actions, if any?

A. Bypass the isolation per GP 8.5 to lower Drywell Pressure

B. Vent the Drywell per OT-101 to lower Drywell Pressure

C. Start 1B Drywell Chiller per S87.1.A to lower Drywell Temperature

D. Spray the Drywell per T-225 to lower Drywell Temperature

Answer: B

Answer Explanation

A Incorrect: Plausible to the candidate who considers bypassing the isolation the correct action but does not recognize that with G532 indicating "Hi/Low" level or Low suction pressure GP-8.5 bypass cannot continue. GP-8.5 requires G532 to be "normal" or Suction pressure to be greater than 35 psig to bypass the isolation.

B Correct: With an Inadvertent Group 8A isolation, Drywell Chillwater is isolated to the Drywell Unit coolers and the drywell will experience a loss of cooling and entry into OT-101. With an unbypassable isolation, the only option to reduce pressure in the drywell is to Vent the drywell per OT-101 Attachment 3, Loss of Drywell Cooling. Initial steps of the attachment direct restarting chillers but with the isolation signal, venting per step 5 of the attachment is correct.

C Incorrect: Plausible to the candidate who believes that the 1B Drywell chiller has a different flowpath from the 1A chiller and would therfore be available with an 8A isolation. In actual fact, the chillers share the same flowpath.

D Incorrect: Plausible to the candidate who believes that spraying the drywell is the only available action. Spraying the drywell is directed out of T-102 but the entry condition of 1.68# drywell presure or 145°F drywell temperature have not been met.

Question 85 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1740796
User-Defined ID:	Q #85 NEW
Lesson Plan Objective:	LGSOPS1540.3
Topic:	(SRO Only) Hi Drywell Temperature
RO Importance:	3.9
SRO Importance:	4.1
K/A Number:	295012 AA2.02

Comments:

General Data			
Level	SRO		
Tier	1		
Group	2		
KA # and Rating	295012 AA2.02		
KA Statement	**Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE:** Drywell pressure		
Cognitive level	High		
Safety Function	5		
10 CFR 55	CFR: 43.5 (b)(5)		
Technical Reference with Revision No:	OT-101	**Rev #:**	37
Justification for Non SRO CFR Link:			
Question History: (i.e. LGS NRC-05, OYS CERT-04)			
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):			
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

86 **ID: 1749051** **Points: 1.00**

**********SRO ONLY**********

Unit 1 is operating at 100% power with the following 1A SRV plant data:

	Baseline Values	Current Values
TE-41-113A	503°F	499°F
TE-41-1N004A	120°F	253°F

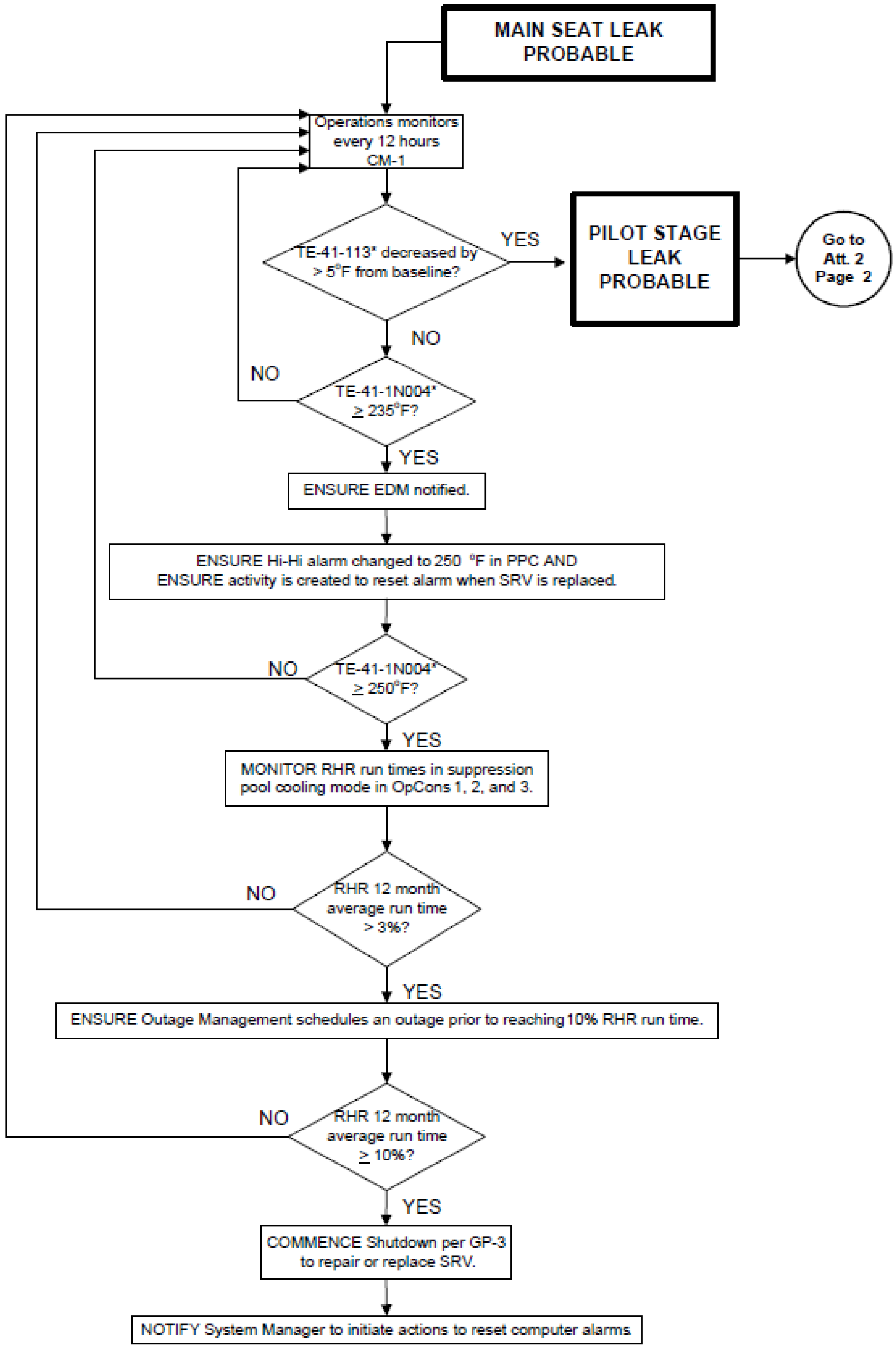
RHR run time total for the past 12 months is 907 hours.

Given the above information, WHICH ONE of the following identifies the required actions, if any, to be directed from RT-6-041-490-1 and the system at risk of water hammer if restarted following a LOCA/LOOP?

(See Next Page)

Attachment 2
Page 3 of 3

SRV LEAKAGE DETERMINATION, MONITORING PROCESS



	RT-6-041-490-1 required action	If in Suppression Pool Cooling Mode, a restart of ______ pumps following a LOCA/LOOP will cause water hammer.
A.	Full power operation may continue per GP-5	RHR
B.	Full power operation may continue per GP-5	RHRSW
C.	Commence shutdown per GP-3	RHR
D.	Commence shutdown per GP-3	RHRSW

Answer: C

Answer Explanation

From the stem the candidate determines that the pilot temperature for the 1A SRV has dropped by 4 °F which is less than the > 5 °F used in RT-6-041-490-1 Attachment 2 for Probable pilot leak. Also from the stem having tailpipe temperature of 253 °F is indicative of probable main seat leak. From Pg 3 of attachment 2, RHR run times of greater than 10% require a Shutdown per GP-3. From the stem RHR run times are 907 hours. 365 days X 24 hrs = 8760 hours in a year. Run time for this question = 907/8760 = 10.35%. With run time greater than 10% the appropriate action is to "Commence Shutdown per GP-3".

The concern with operating Suppression Pool Cooling if a LOCA/LOOP occurs is found in NRC issued Information Notice No. 87-10, "Potential for Water Hammer during Restart of Residual Heat Removal Pumps". The purpose of this notice was to alert the industry of the potential for water hammer in the RHR system during design basis loss of coolant accident (LOCA) coincident with loss of offsite power (LOOP) if the RHR system is aligned to SPC. The power loss while aligned to suppression pool cooling will prevent full closure of HV-051-*F024A(B), allowing portions of the RHR system to drain down to the suppression pool as a result of elevation differences and causing voids in the piping. Upon the subsequent restart and alignment of RHR in the LPCI alignment, the voids may cause a water hammer that could challenge the integrity of the RHR piping.

The concern identified in the above Information Notice is also contained in S51.8.A step 3.10:

3.10 **IF** in Suppression Pool Cooling Mode,
THEN restart of RHR pumps following a LOCA/LOOP will cause water hammer. Using only 1 Loop in Suppression Pool Cooling ensures the other Loop will remain operable following a LOOP event. Pumps are nc considered inoperable in this mode unless a high point vent alarm occu Follow ARC actions to fill and vent to restore pump operability. (Ref. 5.

SRO Only: This question is SRO only because the candidate must assess plant conditions then select the section of the procedure to implement.

A Wrong - Plausible to the candidate that incorrectly determines Run time is less than 10%

B Wrong - Plausible to the candidate that incorrectly determines Run time is less than 10% Also plausible to the candidate that incorrectly recalls the water hammer concern is with RHRSW

C Correct for the reasons described above

D Wrong - plausible to the candidate that incorrectly recalls the water hammer concern is with RHRSW

Question 86 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	0.00
System ID:	1749051
User-Defined ID:	Q #86
Lesson Plan Objective:	LGSOPS0001.12.C
Topic:	SRO ONLY - Leaky SRV
RO Importance:	
SRO Importance:	3.2
K/A Number:	239002 A2.02

Comments:

General Data	
Level	SRO
Tier	2
Group	1
KA # and Rating	239002 A2.02 SRO importance 3.2
KA Statement	239002 SRVs A2.02 - Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Leaky SRV
Cognitive level	High
Safety Function	3 - Reactor Pressure Control
10 CFR 55	43(b)(5)
Technical Reference with Revision No:	ARC-MCR-110 B1 / RT-6-041-490-1 S51.8.A — **Rev #:** 0 / 12 / 049
Justification for Non SRO CFR Link:	N/A
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New
Question Source: (i.e. New, Bank, Modified)	New
Low KA Justification (if required):	N/A
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	RT-6-041-490-1 Attachment 2 PG 3 Imbedded
LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

87	ID: 1749131	Points: 1.00

**********SRO ONLY**********

Plant conditions:

- A loss of the 220 kV and 500 kV Grid occurred 2 hours ago
- D12 DG is running loaded
- D23 DG is running loaded
- All other Unit 1 and Unit 2 DGs have failed to start

D22 Bus needs to be re-energized by cross-tying D23 Bus with D22 using the 201 Safeguard Bus.

WHICH ONE of the following identifies the event procedure requiring entry for the above conditions and the SOURCE BUS 4 kV breaker compartment where the TD-6U Test Device cover is to be removed to allow synchronization of the D23 Bus with the D22 Bus?

Given:

- E-1, Loss of all AC Power (Station Blackout)
- E-10/20, Loss of Offsite Power

	Event Procedure	TD-6U Test Device Cover Removed
A.	E-1	201-D23
B.	E-1	201-D22
C.	E-10/20	201-D23
D.	E-10/20	201-D22

Answer: C

Answer Explanation

From E-10/20

E-10/20 LOSS OF OFFSITE POWER

1.0 CONFIRMING INDICATION

1.1 Loss of voltage for 10 STA on V/103-2(V)
AND 20 STA on V/103-2(V) at 00C660, START-UP.

From E-1

E-1 LOSS OF ALL AC POWER (STATION BLACKOUT)

1.0 CONFIRMING INDICATION

1.1 Loss of 220 KV
AND 500 KV Grid with a failure of all 4 Diesel Generators on a Unit to start (SBO)

1.2 Loss of 220 KV
AND 500 KV Grid with a failure of all 8 Diesel Generators across both Units to start (dual unit SBO)

1.3 Loss of Control Room Instrumentation

1.4 Loss of AC lighting

1.5 Loss of all Plant Ventilation Systems

From E-1 Bases:

The analyzed Station Blackout coping duration is 4 hours. Station blackout coping using an alternate AC (AAC) approach entails a short period of time in an AC-independent state (up to one hour) while the operators initiate power from the AAC source (i.e. spare Diesel Generator capacity on the unaffected unit). Once AAC power is available, the plant transitions to the AAC source and establishes decay heat removal until offsite or Diesel Generator power becomes available.

No Diesel Generators are assumed available in the blacked out unit. For the non-blackout unit, considering the single failure criterion, three Diesel Generators are assumed to be available. For Limerick, there is adequate capacity and capability to power the essential loads in the blackout unit without_requiring any load shedding in the non-blackout unit. Although redundant capability is not available, a fully capable AAC source enables Limerick to attain safe shutdown during an SBO event and recover from the Main Control Room (MCR).

From the stem the candidate determines the entry into E-10/20 is appropriate for the above conditions.

Cross-Tie operation in this case would be performed using E-10/20, Att. 3. Step 2.2.3 directs pulling the cover of the TD-6U device on the SOURCE bus, which will be 201-D23 in this case since D23 is energized.

Clarification of Source Bus vs Target Bus is found in step 2.1.5 of E-10/20:

Question 87 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	0.00
System ID:	1749131
User-Defined ID:	Q #87
Lesson Plan Objective:	LGSOPS0092A.05C
Topic:	SRO - E-10/20 EDG Sync
RO Importance:	0
SRO Importance:	3.6
K/A Number:	264000 A2.05

Comments:

General Data			
Level	SRO		
Tier	2		
Group	1		
KA # and Rating	264000 A2.05 SRO importance 3.6		
KA Statement	264000 EDGs A2.05 - Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Synchronization of the emergency generator with other electrical supplies		
Cognitive level	High		
Safety Function	6 - Electrical		
10 CFR 55	43(b)(5)		
Technical Reference with Revision No:	E-10/20 S32.1.A	**Rev #:**	56 38
Justification for Non SRO CFR Link:	None		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	None		
Question Source: (i.e. New, Bank, Modified)	Modified 1248900		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)	N/A		
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

88	ID: 1749153	Points: 1.00

*************SRO ONLY*************

Unit 2 plant conditions:

- OPCON 5
- Reactor level is 217"
- '2A' RHR is in Shutdown Cooling
- HV-51-2F008, SHUTDOWN COOLING SUCTION OUTBOARD, fails closed and cannot be reopened

WHICH ONE of the following identifies the:

(1) Tech Spec required action, and
(2) the basis for performing that action?

A. (1) Place ADHR in service
(2) To maintain and reduce reactor coolant temperature

B. (1) Place ADHR in service
(2) To assure accurate reactor coolant temperature indication

C. (1) Place 'A' Recirc pump in service
(2) To maintain and reduce reactor coolant temperature

D. (1) Place 'A' Recirc pump in service
(2) To assure accurate reactor coolant temperature indication

Answer: D

Answer Explanation

From the stem the candidate determines that SDC has no suction path (HV-51-2F008 closed). Tech Spec 3.9.11.2 action c.1 requires the verification of reactor coolant circulation by an alternate method.

c. With no RHR shutdown cooling subsystem in operation:

1. Within one (1) hour from discovery of no reactor coolant circulation, and once per 12 hours thereafter, verify reactor coolant circulation by an alternate method; and

2. Once per hour monitor reactor coolant temperature.

The current plant configuration does not support placing ADHR in service due to the need to have the Reactor cavity flooded up.

Placing a Recirc loop in service satisfies Tech Spec 3.9.11.2, ACTION c.1. This only provides for the coolant circulation need to ensure proper reactor coolant temperature monitoring; it does NOT constitute an "alternate Decay Heat Removal method."

Tech Spec 3.9.11 Bases:

An OPERABLE RHR shutdown cooling subsystem consists of one (1) OPERABLE R one (1) heat exchanger, and the associated piping and valves. The requirement having one (1) RHR shutdown cooling subsystem OPERABLE ensures that 1) sufficie cooling capacity is available to remove decay heat and maintain the water in th reactor pressure vessel below 140°F, and 2) sufficient coolant circulation woul available through the reactor core to assure accurate temperature indication. Management of gas voids is important to RHR Shutdown Cooling Subsystem OPERABIL

S51.5.H, RHR Alternate Decay Heat Removal, Prerequisite # 2.7:

2.7 Reactor Cavity is flooded up with Skimmer Surge Tank (SST) level equal to Reactor Cavity level
AND the associated Fuel Pool gates are removed per GP-6.1, Shutdown Operation - Refueling, Core Alteration and Core Off-Loading.

A Wrong - Plausible to the candidate that fails to recall that ADHR can not be placed in service without the Reactor cavity being flooded. Operation of ADHR would support maintaining and reducing reactor coolant temperature.
B Wrong - Plausible to the candidate that fails to recall that ADHR can not be placed in service without the Reactor cavity being flooded. Operation of ADHR would support ensuring adequate monitoring of reactor coolant temperature.
C Wrong - Plausbile to the candidate who belives core ciruclation with recirc is crediddted as an alternate method of decay heat removal.
D Correct for the abover reasons.

Question 88 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1749153
User-Defined ID:	Q #88
Lesson Plan Objective:	LGSOPS0051.24B
Topic:	SRO ONLY - OPCON 5 - Loss of SDC and Tech Spec bases
RO Importance:	
SRO Importance:	4.6
K/A Number:	295021 2.2.39

Comments:	Bank 715236

General Data	
Level	SRO
Tier	2
Group	1
KA # and Rating	205000 2.2.42 SRO Importance 4.6
KA Statement	205000 Shutdown Cooling 2.2.42 - Equipment Control:: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.
Cognitive level	High
Safety Function	4 - Heat Removal from Reactor Core
10 CFR 55	43(b)(2)
Technical Reference with Revision No:	Tech Spec 3.9.11.2 T.S. Bases 3.9.11 — **Rev #:**
Justification for Non SRO CFR Link:	N/A
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Bank
Question Source: (i.e. New, Bank, Modified)	Bank 715236
Low KA Justification (if required):	N/A
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	None
LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

Question 88 Table-Item Links

NRC EXAM - 10 CFR 55 Operators' Licenses

10 CFR 55.43 SRO WRITTEN EXAMINATION

LIMERICK LO Question Category

ILT
NRC
SRO
HI COG

89	ID: 1752990	Points: 1.00

*********SRO ONLY**********

Unit 2 plant conditions are as follows:

- 100% reactor power
- RCIC blocked for turbine governor maintenance

The following sequence of events occur:

10:00 A reactor SCRAM occurs
10:05 Reactor power is 3%
10:05 Reactor water level is +35 inches
10:15 A HPCI Isolation occurs on High Room Temperature due to HVAC isolation

Given: T-249, HPCI/RCIC HIGH AREA TEMPERATURE ISOLATION BYPASS

WHICH ONE of the following identifies the type of HPCI isolation that occurred and the procedure, in addition to T-249, that must be used prior to injecting with HPCI?

	Type of HPCI Isolation	Procedure that must be used prior to injecting with HPCI
A.	Inboard and Outboard	T-270 TERMINATE AND PREVENT INJECTION INTO THE RPV
B.	Inboard and Outboard	T-251 ESTABLISH A HPCI INJECTION FLOW PATH VIA FEEDWATER ONLY
C.	Outboard Only	T-270 TERMINATE AND PREVENT INJECTION INTO THE RPV
D.	Outboard Only	T-251 ESTABLISH A HPCI INJECTION FLOW PATH VIA FEEDWATER ONLY

Answer: B

Answer Explanation

a. Incorrect: Plausible to the SRO candidate who recognizes that with a HPCI high temperature isolation they must determine that HPCI injection requires the performance of T-249 to bypass the High Temp. Isolation and T-251 to inject through feedwater only. T-270 is not peformed because power level is not above 4% per step LQ-7.

b. Correct: A high temperature isolation in the HPCI pump room would cause a Group 4 isolation on both the inboard and outboard isolation channels of NSSSS. To bypass the high temperature isolation, T-249 is performed. HPCI injection is directed in T-117 step LQ-18 through "Feedwater Only" by performing T-251. T-270 is not required per step LQ-7 with power <4%

c. Incorrect: Plausible to the candidate who believes that High Temp in HPCI would only cause an Outboard isolation similar to the operation of the manual isolation pushbutton or outboard isolation of RWCU on High demineralizer inlet temperature. As noted above, a high temperature in the HPCI pump room would cause both an inboard and outboard isolation. T-270 is not peformed because power level is not above 4% per step LQ-7.

d. Incorrect: Plausible to the candidate who believes that High Temp in HPCI would only cause an Outboard isolation similar to the operation of the manual isolation pushbutton or outboard isolation of RWCU on High RGHX inlet temperature.

HPCI room > 180 ISOL both in and outboard T-249 needed to bypass isolation

SRO Only because candidate must recognize that T-270 not required per T-117 LQ-7. This is an assessment of plant conditions which will direct the SRO to chose a strategy for HPCI use.

Question 89 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1752990
User-Defined ID:	Q #89
Lesson Plan Objective:	LGSOPS2003.04
Topic:	HPCI Hi Temp Strategies
RO Importance:	3.7
SRO Importance:	4.7
K/A Number:	206000 G.2.4.6

Comments:	

General Data	
Level	SRO
Tier	2
Group	1
KA # and Rating	206000 G.2.4.6 3.7 / 4.7
KA Statement	High Pressure Coolant Injection System **Knowledge of EOP mitigation strategies.**
Safety function	N/A
Cognitive level	high
10 CFR 55	43.5(b)(5)
Technical Reference with Revision No:	**Rev #:**
Justification for Non SRO CFR Link:	N/A
Question History: (i.e. LGS NRC-05, OYS CERT-04)	From ILT NRC Exam 2005 2015 CERT
Question Source: (i.e. New, Bank, Modified)	Bank 1242567
Low KA Justification (if required):	N/A
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	none
LORT	
PRA: (i.e. Yes or No or #)	Y
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

90	ID: 1759804	Points: 1.00

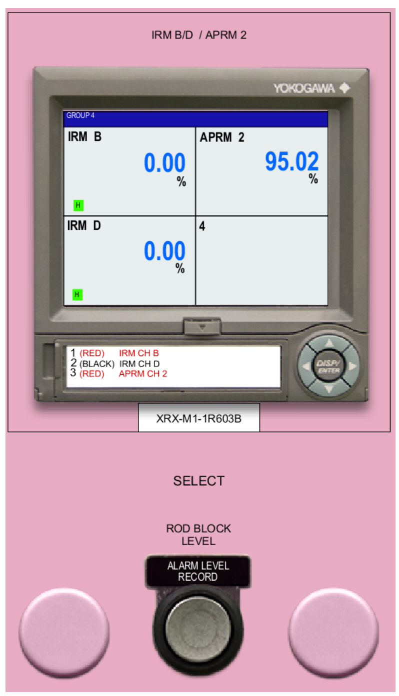
**********SRO ONLY**********

Unit 1 is at 95% power. The Unit 1 RO depresses the "Alarm Level Record" pushbutton (shown below for the #2 APRM) for all 4 Unit 1 APRMs and records the following:

APRM	Reading (%)
1	108.7
2	108.0
3	108.7
4	108.0

Recirc Loop Flows are as follows:

Loop	Flow (kG/min)
1A	36.5
1B	37.2



Given the above information, WHICH ONE of the following correctly identifies the required Tech Spec Actions, if any?

A. No Tech Spec Action required

B. Declare the affected RBM channel inoperable and take the ACTION required by Specification 3.1.4.3

C. Restore the inoperable channel to OPERABLE status within 12 hours or place the inoperable channel in the tripped condition.

D. Place at least one inoperable channel in the tripped condition within one hour.

Answer: C

Answer Explanation

From the stem the candidate determines that the pushbuttons that were pressed by the RO have the APRM recorders swap from displaying APRM power readings to displaying current APRM rod block values.

The equation for calculating the Rod Block Setpoint is found in Tech Spec 3.3.6-2

```
2.  APRM
    a.  Simulated Thermal Power - Upscale:
        -  Two Recirculation Loop Operation          ≤ 0.65 W + 54.3%
                                                     ≤ 108.0% of RATED
                                                     THERMAL POWER
```

Combine this with Lesson Plan LGSOPS0074A definition of W:

c. Each APRM chassis calculates a Total Recirc Drive Flow value (W) ("FLOW (%)") by adding the two flow values received, dividing the summed value by 88,000, and multiplying the quotient by 100.

The calculated Rod Block value is:

0.65(100*(36,500+37,200)/88,000))+54.3% = 108.7%. If the APRMs were working correctly they would output a value of 108.0% due to the second line of Tech Spec Table 3.3.6-2 "clamping" them to 108.0%.

Since 108.7% is above the Table 3.3.6-2 allowable value of 108.4% APRM channels 1 and 3 are INOP and Tech Spec 3.3.6 Action a and b apply. Table 3.3.6-1 identifies the minimum operable channels per trip function for function 2.a at 3. With APRM 1 and 3 INOP there are only 2 remaining channels that are operable and Tech Spec action 61. a applies.

```
ACTION 61   -   With the number of OPERABLE Channels:

                a.  One less than required by the Minimum OPERABLE Channels per Tr
                    Function requirement, restore the inoperable channel to OPERAB
                    status within 12 hours or place the inoperable channel in the
                    tripped condition.
```

A Wrong - plausible to the candidate that correctly calculates the Rod Block value using the equation but fails to recognize that this value is clamped at 108.0%

B Wrong - plausible to the candidate that correctly determines that an action is required but selects action 60. This is easily confused due to the fact that APRMs do provided a power input to the Rod Block Monitors.

C Correct for the above reasons

D Wrong - plausible to the candidate that correctly determines that 2 APRMs are INOP for their Rod Block function but incorrectly determines this constitutes "two or more less than required" due to the fact that only 3 of the 4 APRMs are required.

Question 90 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1759804
User-Defined ID:	Q #90
Lesson Plan Objective:	LLOT0074A.16
Topic:	SRO Only - APRM/LPRM Controls
RO Importance:	
SRO Importance:	4.3
K/A Number:	215005 2.1.31

Comments:

General Data		
Level	SRO	
Tier	2	
Group	1	
KA # and Rating	215005 2.1.31 SRO importance 4.3	
KA Statement	215005 APRM / LPRM 2.1.31 - Conduct of Operations: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.	
Cognitive level	High	
Safety Function	7 - Instrumentation	
10 CFR 55	43(b)(2)	
Technical Reference with Revision No:	Tech Spec 3.3.6 LGSOPS0074A	**Rev #:**
Justification for Non SRO CFR Link:	N/A	
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New	
Question Source: (i.e. New, Bank, Modified)	New	
Low KA Justification (if required):	N/A	
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)		

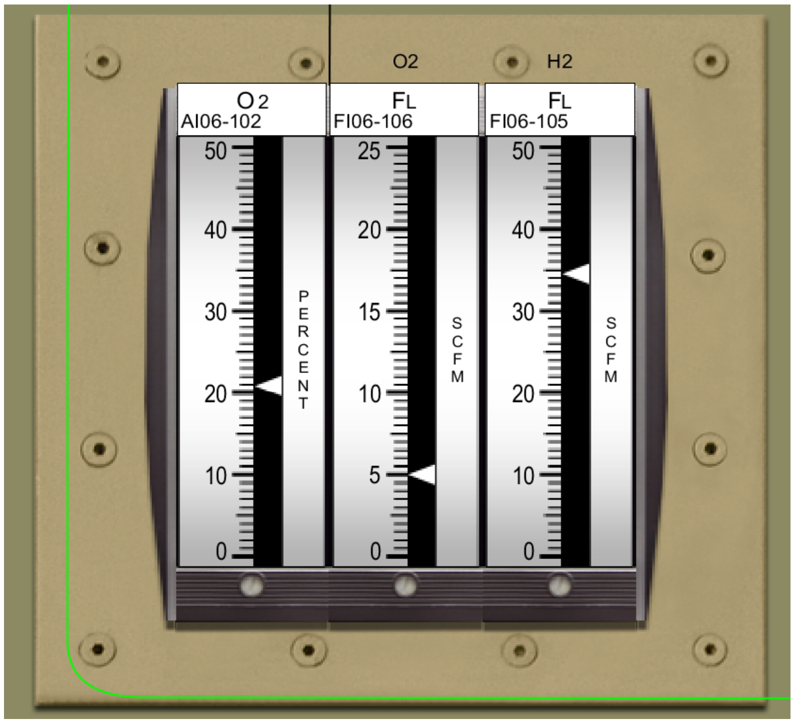
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	Tech Spec 3.3.6 Pages 3/4 3-57 to 3/4 3-60a

LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

91 **ID: 1759803** **Points: 1.00**

Unit 1 is operating at 100% power.

Hydrogen Water Chemistry (HWC) system parameters are as follows:



Given:

- T-103, Secondary Containment Control
- ON-102, Air Ejector Discharge or Main Steam Line High Radiation

WHICH ONE of the following describes:

(1) the plant response, and
(2) the required operator action?

A. (1) A rise in Offgas System temperatures
(2) SCRAM the reactor and close the MSIV's per T-103

B. (1) A rise in Offgas System temperatures
(2) Trip Hydrogen Water Chemistry per ON-102

C. (1) A rise in Main Steam Line radiation
(2) SCRAM the reactor and close the MSIV's per T-103

D. (1) A rise in Main Steam Line radiation
(2) Trip Hydrogen Water Chemistry per ON-102

Answer: D

Answer Explanation
The primary mechanism by which H2 injection causes a temporary rise in MSL and consequently SJAE radiation monitor levels is the transition from oxidizing to reducing conditions in the reactor which cause a temporary reverse in the transport of N13, affecting the radiation monitors. This transition occurs between 5 and 20 SCFM of H2. If a Hydrogen Water Chemistry System malfunction exists, ON-102 (Air Ejector Discharge or Main Steam High Radiation), 2.2 **IF** a Hydrogen Water Chemistry System malfunction exits, **THEN** **TRIP** the Hydrogen Water Chemistry System. THEN **REDUCE** reactor power in accordance with GP-5 Appendix 2, Section 3.1, Reducing Rx Power **AND** Reactor Maneuvering Shutdown Instructions to maintain air ejector discharge radiation level below Hi Hi Alarm setpoint (ARC-MCR-*09 G1) **AND** Main Steam Line radiation below 3x normal full power background. (Hi Hi Alarm Setpoint) **'D is correct: (1) A rise in Main Steam Line radiation; (2) Trip Hydrogen Water Chemistr.** Correct for the reasons described above. 'A' is wrong: (1) A rise in Offgas System temperatures; (2) SCRAM the reactor and close the MSIV's Combustion in the Offgas System is indicated by a sudden drop in Offgas System hydrogen concentration. Plausible to the examinee who confuses the change in HWC H2 injection that is symptomatic of Offgas combustion. Part (2) is taken from T-103 actions for MSL rad hi rad if not caused by a HWC problem 'B' is wrong: (1) A rise in Offgas System temperatures; (2) Trip Hydrogen Water Chemistry. Combustion in the Offgas System is indicated by a sudden drop in Offgas System hydrogen concentration. Plausible to the examinee who confuses the change in HWC H2 injection that is symptomatic of Offgas combustion. 'C' is wrong: (1) A rise in MSL rad; (2) SCRAM and close the MSIV's. Plausible to the candidate who confuses the ON-102 actions with the T-103 actions. MSL rad will rise, however the correct response is to take the action per step 2.3 of ON-102 to lower power as described above.

Question 91 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1759803
User-Defined ID:	Q #91
Lesson Plan Objective:	LGSOPS1550.IL1
Topic:	Predict plant response to increase in HWC H2 injection rate
RO Importance:	2.6
SRO Importance:	2.8
K/A Number:	272000 A2.07

Comments:

General Data	
Level	SRO
Tier	2
Group	2
KA # and Rating	272000 A2.07
KA Statement	A2.07 - Ability to predict the impacts of the following on the RADIATION MONITORING SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Hydrogen injection operation: Plant-Specific
Cognitive level	Lower
Safety Function	9
10 CFR 55	43.(b) (5)
Technical Reference with Revision No:	ON-102 **Rev #:** 30
Justification for Non SRO CFR Link:	
Question History: (i.e. LGS NRC-05, OYS CERT-04)	
Question Source: (i.e. New, Bank, Modified)	Modified from #995211
Low KA Justification (if required):	
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	None
LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

92	ID: 1763353	Points: 1.00

**********SRO ONLY**********

Unit 2 plant conditions are as follows:

- The “2B” Recirc Pump has tripped
- Reactor power is 30%

Subsequently, Power is lost to NXG-II A and NXG-II B for the 2A ASD

Given:

- S43.1.A, Start Up of Recirculation System
- OT-112, Unexpected/Unexplained Change in Core Flow
- OT-200 Appendix 1, Reactor Scram Hard Card

WHICH ONE of the following describes the required action and the basis for this action?

A. Restart a Recirc Pump per S43.1.A to prevent RPV thermal stratification.

B. SCRAM per OT-112 to prevent undue stress on the vessel nozzles and bottom head regions upon startup of an idle loop

C. SCRAM per OT-112 to avoid the reactivity effects of starting a Recirc Pump in natural circulation.

D. SCRAM using OT-200 Appendix 1 to meet the requirements of LCO 3.0.3

Answer: C

Answer Explanation

From the stem the candidate determines that the loss of both NXG controllers to the 2A ASD results in a trip of the 2A Recirc Pump.
From S43.1.F

Display #	Name/Condition	Setting/Indication	Cause and Consequence	Action
1-23	Trip – Power Supply Fault (Low Voltage Power Supply Fault: Power Supply - 1023)	• Alarm: Both power supplies fail. • ASD Fault: Power to both NXG-II DCRs is lost.	• Loss of power supply to the NXG -II DCR, usually because of a loss of power to the chassis (from loss of AC or a failed power supply). • **IF** one power supply fails, **THEN** there is an auto-transfer to the redundant supply. • **IF** both power supplies fail, **THEN** there is an auto-transfer to the backup NXG-II controller (which uses different power supplies). • **IF** power to both NXG-II DCRs is lost, **THEN** ASD will trip.	Operator should: 1. **CHECK** power supply. Have I&C: **CHECK** modulator board.

In this condition (no recirc pumps in operation) OT-112 step 3.1 requires a manual reactor scram.

3.1 **IF** both Reactor Recirc Pumps trip,
THEN manually **SCRAM** the reactor
AND EXIT this procedure
AND ENTER T-100 or T-101 as appropriate.

The basis for the scram is discussed in OT-112 step 6.3

6.3 Prior to OPRM System activation, a scram was required at LGS following entry into natural circulation based on commitments made to NRC Bulletin 88-07 Supplement 1, and was part of the Interim Corrective Actions (ICAs). The basis for the scram was to prevent the occurrence of core THI. Once OPRM System activation occurred, however, LGSs was no longer committed to the ICAs. Even though Technical Specification 3.4.1.1 permits continued plant operation of up to 12 hours following a dual Recirc Pump trip, Exelon has decided, on a fleet-wide basis, to continue to direct that a manual scram be performed any time the plant enters the natural circulation mode, based primarily on the reactivity effects of starting a Recirc Pump while operating in this condition.

A Wrong - Plausible to the candidate that recalls the concerns and cautions associated with starting the second Recirc Pump. Restarting of the first recirc pump is implicitly prohibited based on step 6.3.
B Wrong - Plausible to the candidate that confuses the Tech Spec bases for temperature limits prior to start of an idle recirc pump (3/4.4.1 Bases)
C Correct for the reasons discussed above
D Wrong - plausible to the candidate that does not recall that there is a Tech Spec LCO for operation with no recirc pumps in service (3.4.1.1)

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:	0.00
System ID:	1763353
User-Defined ID:	Q #92
Lesson Plan Objective:	LLOT1540.05
Topic:	SRO Only - Recirc Flow Control Emergency - actions required
RO Importance:	
SRO Importance:	4.4
K/A Number:	202002 2.4.49

Comments:	

General Data			
Level	SRO		
Tier	2		
Group	2		
KA # and Rating	202002 2.4.49 SRO importance 4.4		
KA Statement	202002 Recirculation Flow Control 2.4.49 - Emergency Procedures / Plan: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.		
Cognitive level	High		
Safety Function	1 - Reactivity Control		
10 CFR 55	43.(b)(5)		
Technical Reference with Revision No:	OT-112 S43.1.F	**Rev #:**	5 7 0 4
Justification for Non SRO CFR Link:			
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	N/A		
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

93	ID: 2029740	Points: 1.00

Unit 1 is at 72% with power ascension in progress.

A steam leak results in the following Turbine Enclosure Steam Line Tunnel temperature indications:

- TE25-115C, TURB ENCL MSLT AMB, indicates 193°F up slow @ 1°F/min.
- TE25-115D, TURB ENCL MSLT AMB, indicates 153°F up slow @ 2°F/min.

The following annunciator is in alarm:

- ARC-MCR-107, REACTOR, Window H-5, “DIV 3 STEAM LEAK DET SYS HI TEMP/TROUBLE”

WHICH ONE of the following identifies how long until a Reactor Scram occurs and how long after the RPS actuation NRC notification required?

	Scram Time	Reporting Time
A.	6 minutes	4 hours
B.	6 minutes	8 hours
C.	20 minutes	4 hours
D.	20 minutes	8 hours

Answer: A

Answer Explanation

Justification:

A. **Correct:** Group 1 isolation occurs if A or C AND B or D channels are tripped, and the trip set points are reached when Div 3 Steam Leak Detection System High temp/Trouble annunciator is alarmed. The trip set point for the Turbine Enclosure MSL Leak Detection Channel is 165°F. At the current rate of rise, this would occur in 6 minutes. SAF 1.6 of the Reportability Reference Manual requires NRC notification within 4 hours of an RPs acuation on a critical reactor.

- **RPS Actuation Summary Table**

	Valid RPS Actuation	Valid RPS Actuation	Invalid RPS Actuation	Invalid RPS Actuation
Critical	4 hr. telephone report	60 day LER report	4 hr. telephone report	60 day LER report
Critical (preplanned)	No report	No report	No report	No report
Not Critical	8 hr. telephone report	60 day LER report	No report	No report*
Not Critical (preplanned)	No report	No report	No report	No report

B. **Wrong**: Plausible to the candidate who incorrectly recalls the reporting requirements of SAF 1.6 for a "Not Critical" reactor which is 8 hours.

C. **Wrong**: Plausible if the applicant does not understand that Group 1 isolation occurs at 165° but instead confuses this with the Outboard MSIV room setpoint of 192°F. At the current rate of rise, it would take approx. 20 minutes to reach 192°F.

D. **Wrong**: Plausible if the applicant does not understand that Group 1 isolation occurs at 165° but instead confuses this with the Outboard MSIV room setpoint of 192°F. Also if the candidate incorrectly recalls SAF 1.6

Question 93 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	2029740
User-Defined ID:	Q #93
Lesson Plan Objective:	LGSOPS0072.5G
Topic:	Steam leak in Turbine Enclosure
RO Importance:	3.6
SRO Importance:	4.5
K/A Number:	239001 G2.2.38

Comments:	

General Data	
Level	SRO
Tier	2
Group	2
KA # and Rating	239001 2.2.38 3.6 / 4.5
KA Statement	**Main and Reheat Steam**: Equipment Control: Knowledge of conditions and limitations in the facility license.
Cognitive level	High
Safety Function	3
10 CFR 55	43 (b)(1)
Technical Reference with Revision No:	ARC-MCR107 H-5, I-5 T-101 — **Rev #:**
Justification for Non SRO CFR Link:	
Question History: (i.e. LGS NRC-05, OYS CERT-04)	
Question Source: (i.e. New, Bank, Modified)	New
Low KA Justification (if required):	
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	None
LORT	
PRA: (i.e. Yes or No or #)	
LORT Question Section: (i.e, A-Systems or B-Procedures)	
Comments	

#1097850

94	ID: 1769592	Points: 1.00

**********SRO ONLY**********

WHICH ONE of the following correctly completes the below statements regarding Tech Spec system limits associated with surveillance test frequencies?

Each Surveillance Requirement shall be performed within the specified surveillance time interval with a maximum allowable extension not to exceed ___(1)___ of the surveillance interval.

If it is discovered that a Surveillance was not performed within its specified Surveillance time interval and allowed extension, then compliance with the requirement to declare the LCO not met may be delayed, up to ___(2)___ or up to the limit of the specified Surveillance time interval, whichever is greater.

A. (1) 10%
(2) 12 Hours

B. (1) 10%
(2) 24 Hours

C. (1) 25%
(2) 12 Hours

D. (1) 25%
(2) 24 Hours

Answer: D

Answer Explanation

The answer to both statements is found in Tech Spec sections 4.0.2 and 4.0.3

```
4.0.2  Each Surveillance Requirement shall be performed within the specified
surveillance time interval with a maximum allowable extension not to exceed 25%
of the surveillance interval.

4.0.3  If it is discovered that a Surveillance was not performed within its
specified Surveillance time interval and allowed extension per Specification
4.0.2, then compliance with the requirement to declare the Limiting Condition for
Operation not met may be delayed, from the time of discovery, up to 24 hours or
up to the limit of the specified Surveillance time interval, whichever is
greater.  This delay period is permitted to allow performance of the
Surveillance.  A risk evaluation shall be performed for any Surveillance delayed
greater than 24 hours and the risk impact shall be managed.
```

A Wrong - plausible to the candidate that incorrectly recalls both the value of the allowable grace and the missed surveillance extension time. The values of 10% and 12 hours appear multiple times in Tech Specs for various items.

B Wrong - plausible to the candidate that incorrectly recalls both the value of the allowable grace. The value of 10% appears multiple times in Tech Specs for various items.

C Wrong - plausible to the candidate that incorrectly recalls the missed surveillance extension time. The value of 12 hours appears multiple times in Tech Specs for various items.

D Correct for the above reasons

Question 94 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1769592
User-Defined ID:	Q #94
Lesson Plan Objective:	LGSOPS1800.6
Topic:	SRO Only - Explain and apply system limits and precautions
RO Importance:	
SRO Importance:	4.0
K/A Number:	LGSOPS1800.6

Comments:

General Data		
Level	SRO	
Tier	3	
Group		
KA # and Rating	2.1.32	
KA Statement	2.1.32 - Ability to explain and apply all system limits and precautions.	
Cognitive level	Low	
Safety Function	N/A	
10 CFR 55	43(b)(2)	
Technical Reference with Revision No:	Tech Spec 4.0.2 and 4.0.3	**Rev #:**
Justification for Non SRO CFR Link:	N/A	
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New	
Question Source: (i.e. New, Bank, Modified)	New	
Low KA Justification (if required):		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)		
ILT		
Supplied Ref (If appropriate): (i.e. ABN-##)	None	
LORT		
PRA: (i.e. Yes or No or #)		
LORT Question Section: (i.e, A-Systems or B-Procedures)		
Comments		

95	ID: 1770570	Points: 1.00

**********SRO ONLY**********

Given the following:

Unit 1 is operating at 100% power

- A MCR annunciator has alarmed 12 times within the first two hours of the shift
- It has been determined that the alarm is a nuisance and that placing the annunciator mode switch in MANUAL will prevent further nuisance alarms.
- I&C has concluded that lifting a lead to the annunciator will allow the annunciator mode switch to be restored to AUTO.

Per the guidance of OP-AA-103-102 “Watch-Standing Practices” AND OP-LG-103-102-1001 "Alarms And Indications":

The ___(1)___ must authorize placing the annunciator mode switch in MANUAL.
Prior to lifting the lead to the annunciator engineering must complete ___(2)___.

A. (1) CRS
(2) Technical Evaluation/TCCP (CC-AA-112)

B. (1) CRS
(2) MAINTENANCE ALTERATIONS LOG (MA-AA-716-100-F-01)

C. (1) Shift Manager
(2) Technical Evaluation/TCCP (CC-AA-112)

D. (1) Shift Manager
(2) MAINTENANCE ALTERATIONS LOG (MA-AA-716-100-F-01)

Answer: A

Answer Explanation

The question is asking requirement of OP-AA-103-102, Watch-Standing Practices and OP-LG-103-102-1001, Alarms and Indications.

From OP-AA-103-102 step 4.5.5:

– If operational or maintenance corrective actions are ineffective, consideration should be given for **UTILIZATION** of the annunciator mode switch in the manual mode for the duration of the nuisance input or disabling the nuisance input, under the following conditions:
    - **OBTAIN** Unit Supervisor permission.

From OP-LG-103-102-1001:

2.4. To eliminate the distraction of a nuisance annunciator the CRS may direc positioning the annunciator mode switch to MANUAL for the duration of th nuisance signal.

AND

2.8. A technical evaluation/TCCP from engineering is required prior to lifting any leads defeat annunciator inputs.

The MA-AA-716-100 Distractor is wrong but plausible due to this process used at Limerick for some plant alterations.

From MA-AA-716-100:

3.2.2. **ENSURE** proposed Maintenance Alteration Process excluded from governance of CC-AA-112. **If** determined that proposed alteration **not** excluded, **then COMPLY** with requirements of CC-AA-112.

The shift manager distractor is wrong but plausible due the following step in CC-AA-112:

3.10. **Operations Shift Manager** - The Shift Manager has overall responsibility for contr TCCs during the assigned shift.

A Correct for the above reasons
B Wrong - plausible to the candidate that recalls a different temporary control process
C Wrong - plausible to the candidate that believes that the shift manager is required to authorize the annunciator mode switch repositioning
D Wrong - plausible to the candidate that believes that the shift manager is required to authorize the annunciator mode switch repositioning and plausible to the candidate that recalls a different temporary control process

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:	0.00
System ID:	1770570
User-Defined ID:	Q #95 NEW
Lesson Plan Objective:	LOT1572 COND. OF OPS
Topic:	SRO Only - INOP annunciator
RO Importance:	
SRO Importance:	3.3
K/A Number:	2.2.43

Comments:

General Data			
Level	SRO		
Tier	3		
Group	n/a		
KA # and Rating	2.2.43 SRO importance 3.3		
KA Statement	**Knowledge of the process used to track inoperable alarms.**		
Cognitive level	lower		
Safety Function	n/a		
10 CFR 55	43(b)(5)		
Technical Reference with Revision No:	OP-LG-103-102-1001 MA-AA-716-100 OP-AA-103-102	**Rev #:**	9 13 16
Justification for Non SRO CFR Link:	n/a		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	n/a		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	n/a		
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

96	ID: 2036649	Points: 1.00

************SRO ONLY*************

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

- The Containment Leak Detection Radiation Monitor has failed downscale
- The Drywell Sump Monitoring System has failed its monthly Channel Functional Test

WHICH ONE of the following identifies the most limiting Tech Spec required action?

A. Perform primary containment atmospheric grab sampling/analysis every 12 hours and restore the Containment Leak Detector Radiation Monitor to OPERABLE status within 30 days.

B. Perform primary containment atmospheric grab sampling/analysis every 12 hours and monitor Reactor Coolant System leakage by administrative means once per 12 hours and restore either the Containment Leak Detector Radiation Monitor or the Drywell Sump Monitoring System to OPERABLE status within 7 days.

C. Be in HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

D. Restore the Drywell Sump Monitoring System to OPERABLE within 30 days and increase monitoring of Drywell Unit Cooler flow rate to once every 8 hours.

Answer: C

Answer Explanation

Provided reference(s): U/2 Tech Spec 3.4.3.1

The Containment Leak Detection Radiation Monitor is used to aid the MCR staff in identifying a potential leak in the drywell. As such when it is not working (INOP) Tech Specs requires entry into an LCO. Given the conditions in the stem the candidate determines that the Containment Leak Detection Radiation Monitor is INOP as a result of failing downscale. Additionally the candidate determines that the drywell sump monitoring system is also INOP due to failing its channel functional test.

Refer to Tech Spec 3.4.3.1, where 3.4.3.1.a identifies the inop Containment Leak Detection Rad Monitor, and 3.4.3.1.b identifies the inop Drywell Floor Drain Tank Monitor. ACTIONs A, B, C, D, E, and F considers various combinations of inoperable portions of this LCO, except for the combined rad monitor and drywell sump monitor. Therefore, ACTION 'G' applies, which requires the plant to be in HOT SHUTDOWN within 12 hours, etc.

A Wrong - Although an accurate LCO for the conditions in the stem, it is not the most limiting LCO. Plausible to the candidate that fails to recognize that action G applies and is the most limiting.

B Wrong - plausible to the candidate the mis-interprets action F and applies it to the stem conditions, and plausible to the candidate that fails to recognize that action G applies and is the most limiting.

C Correct for the above reasons.

D Wrong - Although an accurate LCO for the conditions in the stem, it is not the most limiting LCO. Plausible to the candidate that fails to recognize that action G applies and is the most limiting.

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:	0.00
System ID:	2036649
User-Defined ID:	Q #96
Lesson Plan Objective:	LLOT0135.7B
Topic:	(SRO Only) - Tech Spec - Determine Action for INOP Containment Leak Detection Systems
RO Importance:	2.9
SRO Importance:	2.9
K/A Number:	G2.3.5

Comments:	

General Data			
Level	SRO		
Tier	3		
Group			
KA # and Rating	G2.3.5 2.9 / 2.9		
KA Statement	**Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.**		
Cognitive level	High		
Safety Function	9		
10 CFR 55	43(b)(2)		
Technical Reference with Revision No:	T.S. 3.4.3.1	**Rev #:**	
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	556506		
Question Source: (i.e. New, Bank, Modified)	Bank 556506		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor "b" to make plausible based on OTPS review)	Removed reference to 3 days into the 30-day LCO from the stem. Re-wrote choice B to make it more plausible.		
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	U/1 T.S. 3.4.3.1 Pages 4-8 to 4-8a		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

97	ID: 1770590	Points: 1.00

**********SRO ONLY**********

Unit 1 was operating at 100% power

- Drywell pressure began to rise due to an unisolable leak from a Reactor Recirc Pump Seal
- Annunciator 107-F2, DRYWELL HI/LO PRESS alarms
- OT-112, "Unexpected/Unexplained Change in Core Flow" is entered
- At 1.3 psig Drywell pressure the crew performed a GP-4 "Rapid Plant Shutdown"

Then:

- Drywell pressure is 1.72 psig up slow
- Drywell temperature is 148°F up slow

Given:

- OT-101, Hi Drywell Pressure
- T-101, RPV Control
- T-102, Primary Containment Control

Which one of the following is correct regarding additional procedures to be used to mitigate the event?

A. T-101 and T-102 must be entered. All AOPs must be exited. Response to the event is as directed by the TRIPs

B. T-101 and T-102 must be entered. OT-101 may also be entered as determined by the CRS. If a conflict arises, the actions of applicable OTs take precedence.

C. OT-101 must be entered. T-101 and T-102 must be entered and executed concurrently. If a conflict arises, the actions of the TRIPs take precedence

D. OT-101 must be entered. T-101 and T-102 are entered but execution/actions are delayed until intial actions of AOPs are completed.

Answer: C

Answer Explanation

A Wrong: Plausible to the candidate who confuses the guidance for EOP completion. With the given condition, T-101 and T-102 entry are appropriate. However, during execution of EOPs, AOPs and Off Normal Procedures (ONs, OTs, SEs etc.) are not exited. They would be performed concurrently

B Wrong: Plausible to the candidate who incorrectly assumes that if an ON or OT is entered, the actions of the ON or OT takes precedence since it is written for the specific transient whereas the EOP is more generic in design.

C Correct: During a transient condition, the EOPs are the most important procedure. ONs and OTs are subordinate to the EOPs. While both EOPs and AOPs can proceed in parallel, in the event of conflicting direction, the EOP takes precedence.

D Wrong: Plausible to the candidate who believes that all actions in an AOP must be completed first prior to EOP execution.

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:	0.00
System ID:	1770590
User-Defined ID:	Q #97
Lesson Plan Objective:	LGSOPS1560.02A
Topic:	Use of AOPs and EOPs
RO Importance:	3.8
SRO Importance:	4.5
K/A Number:	2.4.8

Comments:

General Data			
Level	SRO		
Tier	3		
Group			
KA # and Rating	2.4.8 3.8 / 4.5		
KA Statement	**Knowledge of how abnormal operating procedures are used in conjunction with EOPs**		
Cognitive level	L		
Safety Function			
10 CFR 55	43(b)(5)		
Technical Reference with Revision No:	OP-LG-103-102-1002	**Rev #:**	26
Justification for Non SRO CFR Link:			
Question History: (i.e. LGS NRC-05, OYS CERT-04)			
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):			
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

98	ID: 1795570	Points: 1.00

***********SRO ONLY***********

Unit 2 LOCA is in progress, with the following:

- A Hurricane is in progress with sustained winds of 100 mph coming from 120 degrees.
- No emergency classification has been made
- No turnover of command and control has taken place
- North Stack effluent reading is 2.1 E +7 µCi/sec
- Offsite dose assessment is 5125 mRem CDE Thyroid
- State has notified LGS of impediments to evacuation

As a result of the radiation hazard posed by the above conditions, WHICH ONE of the following identifies the required PAR, if any?

A. No PAR Required

B. Shelter 2 mile radius and shelter 2-5 miles WSW / W / WNW / NW / NNW

C. Evacuate 2 mile radius and evacuate 2-5 miles WSW / W / WNW / NW / NNW

D. Evacuate 2 mile radius, evacuate 2-5 miles WSW / W / WNW / NW / NNW, and shelter 5-10 miles WSW / W / WNW / NW / NNW

Answer: B

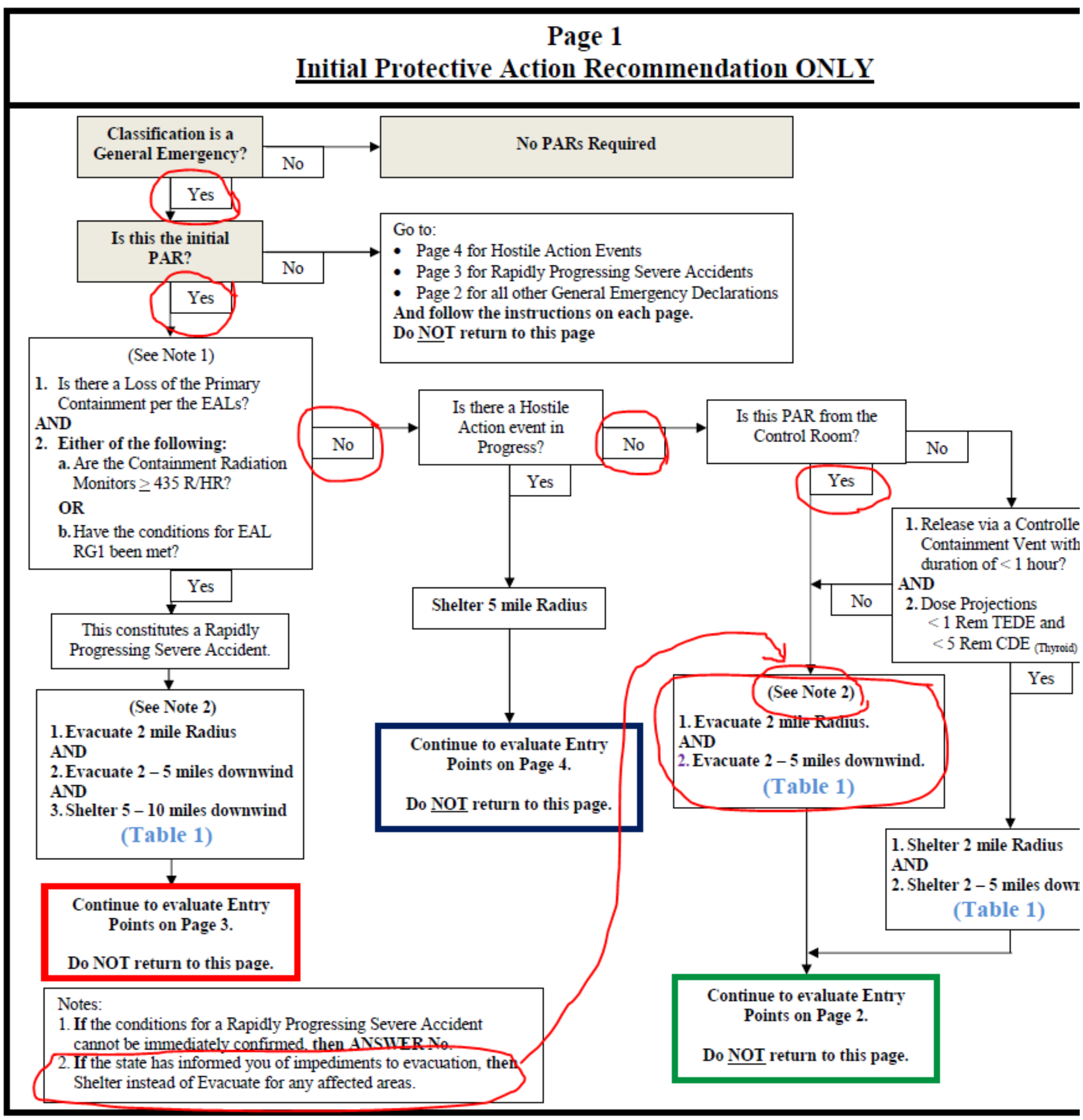
Answer Explanation

Provide EP-AA-111-F-11 and EP-AA-1008 Addendum 3

From the stem conditions the student should make a General Emergency declaration based on the 5125 mRem CDE Thyroid (the proceeding note refers the student to use the dose assessment instead of the north stack readings) meeting the threshold for RG1 - 2.b.

2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**:
   a. **> 1000 mRem TEDE**
   OR
   b. **> 5000 mRem CDE Thyroid**

With no emergency classification and no change of command and control the student is expected to reach the box to Evacuate 2 mile radius and evacuate 2-5 miles downwind on Page 1 of EP-AA-111-F-11.



However, with a hurricane in progress the student should realize that it is prudent to shelter (see note 2) vice evacuate.

Question 98 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	1795570
User-Defined ID:	Q #98
Lesson Plan Objective:	LIMERICK LOT 7340
Topic:	SRO ONLY - Determine PAR due to radiation hazards
RO Importance:	
SRO Importance:	3.8
K/A Number:	2.3.14

Comments:

General Data			
Level	SRO		
Tier	3		
Group	N/A		
KA # and Rating	2.3.14 SRO Importance 3.8		
KA Statement	2.3.14 Knowledge of radiation or containment hazards that may arise during normal, abnormal, or emergency conditions or activities.		
Cognitive level	High		
Safety Function	N/A		
10 CFR 55	43.(b)(4)		
Technical Reference with Revision No:	EP-AA-111-F-11 EP-AA-1008 Add 3	**Rev #:**	A 2
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	LGS 2010 CERT		
Question Source: (i.e. New, Bank, Modified)	Bank 562492		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)	Replaced wrong wind direction disctractor to make more plausible.		
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	EP-AA-111-F-11 EP-AA-1008 Addendum 3 Page 13		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

Question 98 Table-Item Links

NRC EXAM - 10 CFR 55 Operators' Licenses

10 CFR 55.43 SRO WRITTEN EXAMINATION

LIMERICK LO Question Category

ILT
NRC
SRO
HI COG

99 **ID: 1795598** **Points: 1.00**

**********SRO ONLY**********

In reference to HU3, FIRE potentially degrading the level of safety of the plant, WHICH ONE of the SE-8, FIRE, response results listed below would require declaration of an Unusual Event?

HU3 FIRE potentially degrading the level of safety of the plant. 1 2 3 4 5 D

Emergency Action Level (EAL):

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

Escalation of the emergency classification level would be via IC CA2 or MA5.

1. A FIRE in **ANY** Table H2 area is **not** extinguished in **< 15-minutes** of **ANY** of the following FIRE detection indications:
   - Report from the field (i.e., visual observation)
   - Receipt of multiple (more than 1) fire alarms or indications
   - Field verification of a single fire alarm

OR

2. a. Receipt of a single fire alarm in **ANY** Table H2 area (i.e., no other indications of a FIRE).

   AND

   b. The existence of a FIRE is **not** verified in **< 30 minutes** of alarm receipt.

OR

3. A FIRE within the plant PROTECTED AREA not extinguished in **< 60-minutes** of the initial report, alarm or indication.

OR

4. A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.

A. A fire on the 3rd floor of the SSB extinguished by the LGS Fire Brigade 51 minutes following initial fire alarm

B. A fire on the 2nd floor of the Kemper building extinguished by the LGS Fire Brigade and Linfield Fire Department 27 minutes following initial fire alarm.

C. A fire alarm only, in the Spray Pond Pump House that took 16 minutes for the FBL to confirm no smoke/no fire.

D. A fire in the Unit 2 EDG corridor extinguished by the FBL using a handheld fire extinguisher 12 minutes following initial alarm. The fire was contained to a dirty rag barrel.

Answer: B

Answer Explanation
All answer options require entry into SE-8, Fire. SE-8 contains actions to be implemented upon the discovery of a fire, among those actions is to document start times for potential Emergency declaration clocks and notifications to the Shift Manager and WCS. This question is asking which response and results would required declaration of an Unusual Event. With only HU3 section of EP-AA-1008 Addendum 3 being provided, the candidate must recall what areas are part of the Table H2 Vital Areas and the definition of the plant PROTECTED AREA.

Table H2 Vital Areas
• Reactor Enclosure (when inerted the Drywell is exempt) • Control Enclosure • Diesel Generator Enclosure • Spray Pond Pump House / Spray Network

PROTECTED AREA: An area that normally encompasses all controlled areas within the security protected area fence.

A Wrong - Threshold 3 is being approached but is not met until the fire burns for 60 minutes or more

B Correct - Threshold 4 has been met due to offsite fire response agency assistance

C Wrong - Threshold 2 is being approached but is not met due to FBL confirmation of no fire prior to

D Wrong - Threshold 1 is being approached but is not met until the fire burns for 15 minutes or more

Question 99 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1795598
User-Defined ID:	Q #99
Lesson Plan Objective:	EAL OBJECTIVES
Topic:	SRO Only - Fire in the plant procedures - EAL
RO Importance:	
SRO Importance:	3.9
K/A Number:	2.4.27
Comments:	

General Data			
Level	SRO		
Tier	3		
Group	N/A		
KA # and Rating	2.4.27 SRO Importance 3.9		
KA Statement	2.4.27 - Knowledge of "fire in the plant" procedures.		
Cognitive level	Low		
Safety Function	N/A		
10 CFR 55	43(b)(5)		
Technical Reference with Revision No:	EP-AA-1008 Add. 3	**Rev #:**	2
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	New		
Question Source: (i.e. New, Bank, Modified)	New		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

100	ID: 1795594	Points: 1.00

**********SRO ONLY*********

A Unit 1 activity has been determined to be a 'Special Test or Evolution' (STE), per OP-AA-108-110.

The STE is to be performed by the on-watch, Main Control Room personnel.

WHICH ONE of the following personnel can be assigned to be the 'STE Coordinator'?

A. Shift Manager

B. Unit 1 CRS

C. PRO

D. Responsible System Manager

Answer: D

Answer Explanation	
Refer to OP-AA-108-110, section 4.3 CAUTION, which states that the STE Coordinator is NOT to be an active performer of the test. Stem conditions indicate that U1 MCR personnel are the "performers"; therefore, given the answer choices, only the Responsible System Manager is eligible to act as Coordinator.	
ANSWER: Responsible System Manager	
A	Wrong - plausible if the candidate confuses the requirements of step 3.4 (Operations Shift Management responsibilities) with those of The STE Coordinator
B	Wrong - plausible if the candidate confuses the requirements of step 3.4 (Operations Shift Management responsibilities) with those of The STE Coordinator
C	Wrong - Plausible if the candidate fails to recall the content of the CAUTION described above
D	Correct for the above reasons

Question 100 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	0.00
System ID:	1795594
User-Defined ID:	Q #100
Lesson Plan Objective:	LGSOPS2005A.4
Topic:	SRO ONLY - Recall who can act as Coordinator for a Special Test or Evolution
RO Importance:	
SRO Importance:	3.6
K/A Number:	2.2.7

Comments:

General Data			
Level	SRO		
Tier	3		
Group	N/A		
KA # and Rating	2.2.7 SRO Importance 3.6		
KA Statement	2.2.7 Knowledge of the process for conducting special or infrequent tests.		
Cognitive level	Low		
Safety Function	N/A		
10 CFR 55	43(b)(5)		
Technical Reference with Revision No:	OP-AA-108-110	**Rev #:**	3
Justification for Non SRO CFR Link:	N/A		
Question History: (i.e. LGS NRC-05, OYS CERT-04)	Bank		
Question Source: (i.e. New, Bank, Modified)	Bank 985682		
Low KA Justification (if required):	N/A		
Revision History: Revision History: (i.e. Modified distractor “b” to make plausible based on OTPS review)			
ILT			
Supplied Ref (If appropriate): (i.e. ABN-##)	None		
LORT			
PRA: (i.e. Yes or No or #)			
LORT Question Section: (i.e, A-Systems or B-Procedures)			
Comments			

Question 100 Table-Item Links

LIMERICK LO Question Category

ILT
NRC
SRO
HI COG

NRC EXAM - 10 CFR 55 Operators' Licenses

10 CFR 55.43 SRO WRITTEN EXAMINATION

CFR: 43.3 Facility licensee procedures required to obtain authority for design and operating changes in the facility.