Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Pressurizer Vapor Space Accident: Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: Criteria for throttling high-pressure injection after a small LOCA	**Tier**	**1**	[redacted]	
	Group	**1**	[redacted]	
	K/A	**008 AA2.23**		
	IR	**3.6**	[redacted]	

Question 1

Given the following conditions:

- Unit 1 was tripped from 100% power due to a Pressurizer Safety lifting and sticking open
- SPTAs have been performed and the CRS has entered 40EP-9EO03, LOCA
- The RCS is 35°F subcooled and stable
- Indicated Pressurizer level is 90% and slowly rising
- Both SG levels are 15% NR and slowly rising, being fed from AFB-P01
- QSPDS shows two HJTCs are uncovered in the vessel head (41% level in the head)
- Containment temperature is 150°F and slowly rising
- Containment High Range Area Radiation Monitors, RU-148 and RU-149, indicate 6.5×10^2 mrR/hr and slowly rising

Per Appendix 2, HPSI Throttle Criteria, HPSI throttle criteria…

A. IS currently satisfied

B. is NOT satisfied due to voiding in the vessel head

C. is NOT satisfied due to insufficient level in the SGs

D. is NOT satisfied due to insufficient RCS subcooling

Proposed Answer:	A

Explanations:	
A.	Correct.
B.	Plausible as there is voiding in the upper head, however level in the upper head needs to be 16% or more to throttle HPSI, therefore with level in the upper head is at 67%, inventory is sufficient.
C.	Plausible as level in the SGs is 30% less than the normal post trip SG level control band, however since level is being restored, it meets HPSI throttle criteria.
D.	Plausible since subcooling would be insufficient if containment conditions were harsh, and it is plausible that containment conditions are harsh since temperature and radiation levels are significantly higher than normal levels, however containment temperature and radiation levels are below the threshold for declaring harsh containment conditions.

Question Source:		New	
	X	Bank	
		Modified	
	X	Previous NRC Exam	2016 NRC Exam Q2

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	Given conditions of safety injection flow following a transient, analyze whether it is permissible to throttle HPSI flow	

Technical Reference:	40EP-9EO03, LOCA

PALO VERDE NUCLEAR GENERATING STATION LOSS OF COOLANT ACCIDENT	40EP-9EO03 Revision 44 Page 3 of 79

3.0 INSTRUCTIONS/CONTINGENCY ACTIONS

INSTRUCTIONS | CONTINGENCY ACTIONS

-- NOTE --

Harsh conditions are containment temperature greater than 170°F or containment radiation level greater than 10^8 mR/hr. Harsh containment values are placed in brackets next to the normal setpoint or band.

Technical Reference:	Appendix 2, Figures

PALO VERDE PROCEDURE	40EP-9EO10-002 Revision 3
APPENDIX 2: FIGURES	Page 4 of 6

SI THROTTLE CRITERIA

CAUTION

Throttling HPSI injection valves will cause erosion damage to downstream piping.

HPSI THROTTLE CRITERIA

- At least one HPSI Pump is operating
- RCS is greater than or equal to 24°F [44°F] subcooled
- Pressurizer level is greater than 10% [15%] and **NOT** lowering
- At least one Steam Generator is available for RCS heat removal with level being maintained within or being restored to 45 - 60% NR [45 - 60% NR]
- RVLMS indicates RVUH level is greater than or equal to 16%
- **IF** the Functional Recovery procedure is in use,
  THEN ensure HPSI Pump(s) are **NOT** being used to meet an RC success path

LPSI THROTTLE CRITERIA

- Pressurizer pressure is greater than 220 psia [220 psia] and is being controlled

Technical Reference:	QSPDS Tech Manual

The following table demonstrates the range of actual level vs. indicated level on QSPDS. Note that the indicated level is half way between the minimum and the maximum - all you know is that you are above the maximum of the next lower detector.

Table 3 - 1

Detectors Uncovered	QSPDS Indicated Level (%)		VOID SIZE (FT³)		
	Head	Plenum	Minimum	Indicated	Maximum
None	100	100	0	0	230
1	67	100	230	500	810
1-2	41	100	810	1083	1354
1-3	16	100	1354	1627	1898
1-4	0	100	1898	1967	2039
1-5	0	73	2039	2100	2161
1-6	0	47	2161	2222	2284
1-7	0	21	2284	2345	2406
1-8	0	0	2406	2441	--

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Small Break LOCA: Knowledge of the operational implications of the following concepts as they apply to the small break LOCA: Use of steam tables	Tier	1		
	Group	1		
	K/A	009 EK1.02		
	IR	3.5		

Question 2

Given the following conditions:

- Unit 2 was tripped from 100% power due to an RCS leak
- SIAS and CIAS were manually initiated following the Reactor trip
- RCS pressure is 1800 psia and slowly lowering
- RCS Thot is 565°F and stable
- RCS Tcold is 564°F and stable
- Containment pressure is 1.2 psig and slowly rising
- SPTAs have just been completed

Assuming all applicable contingency actions have been taken, which of the following describes the current status of RCS subcooling and RCP operation?

A. RCS subcooling is sufficient for RCP operation and all 4 RCPs should be running

B. RCS subcooling is sufficient for RCP operation, however only 2 RCPs should be running due to RCS pressure

C. RCS subcooling is sufficient for RCP operation, however all 4 RCPs should be stopped due to SIAS and CIAS being actuated

D. RCS subcooling is insufficient for RCP operation and all 4 RCPs should be stopped

Proposed Answer:	B

Explanations:	
A.	Plausible since the RCS is still subcooled, however if RCS pressure drops below the SIAS setpoint (1837 psia), one RCP in each loop should be stopped.
B.	Correct.
C.	Plausible since RCPs are required to be stopped following a containment isolation due to the loss of NC flow to the RCPs, however the NC CIVs close on a phase B CI signal (CSAS), not a phase A signal (CIAS)
D.	Plausible since subcooling is lower than expected post-trip, however the 24°F subcooling limit for RCP operation has not yet been reached.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		**Memory or Fundamental Knowledge**
	X	**Comprehension or Analysis**

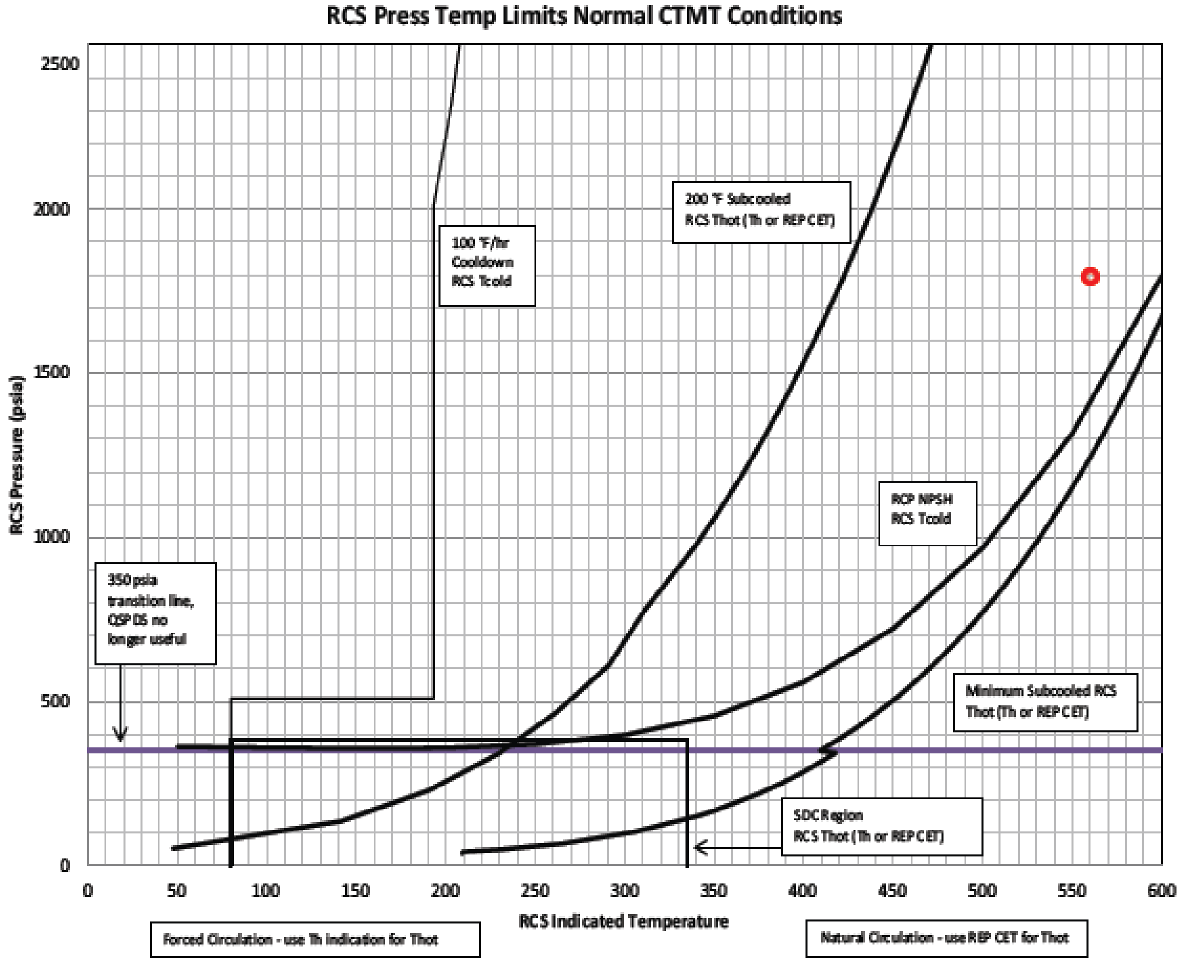
Level of Difficultly:	**3**	
10CFR55.41:	**10**	
Reference Provided:	**N**	
Learning Objective:	**Given RCS pressure and temperature during performance of an EOP, analyze these conditions to decide if the RCPs can be operated per the applicable EOP.**	

Technical Reference:	40EP-9EO01, SPTAs

INSTRUCTIONS	CONTINGENCY ACTIONS
5. Determine that RCS Pressure Control acceptance criteria are met by **BOTH** of the following: • Pressurizer pressure is 1837 - 2285 psia • Pressurizer pressure is trending as expected to 2225 - 2275 psia	5.1 Restore and maintain pressurizer pressure to the normal control band by **ANY** of the following: • Operation of PPCS • Manual operation of pressurizer heaters and spray valves 5.2 **IF** pressurizer pressure drops to the SIAS setpoint, **THEN** ensure that SIAS is actuated. 5.3 **IF** pressurizer pressure remains below the SIAS setpoint, **THEN** stop **ONE** RCP in each loop. 5.4 **IF** pressurizer pressure drops below the RCP NPSH limits, **THEN** stop all RCPs. REFER TO Appendix 2, Figures.
6. Determine that Core Heat Removal acceptance criteria are met by **ALL** of the following: • At least one RCP is operating • Loop ΔT is less than 10°F • RCS is 24°F or more subcooled	

Technical Reference:	

The spot indicated below is above the minimum NPSH for RCP operations (~564F and 1800 psia)



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Large Break LOCA: Ability to operate and monitor the following as they apply to a Large Break LOCA: Core flood tank initiation	Tier	1		
	Group	1		
	K/A	011 EA1.09		
	IR	4.3		

Question 3

Given the following conditions:

- Unit 2 was just manually tripped from 100% power due to a Large Break LOCA
- RCS pressure is 2000 psia and lowering
- Containment pressure is 2.0 psig and rising
- NO additional manual actions have been taken by the crew

SIT Outlet MOVs are currently ___(1)___ and the SITs will BEGIN injecting into the RCS when RCS pressure lowers to approximately ___(2)___ psia.

A. (1) open
(2) 410

B. (1) open
(2) 600

C. (1) closed
(2) 410

D. (1) closed
(2) 600

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible since the SIT outlet valves do get an open signal at 410 psia (when pressure is rising), however injection starts at ~ 600 psia.
B.	Correct.
C.	First part is plausible since SIAS has not yet actuated and SIT outlet valves get an open signal when SIAS actuates, however the SIT outlet valves are maintained open when in MODE 1 so they would already be open. Second part is plausible since the SIT outlet valves do get an open signal at 410 psia (when pressure is rising), however injection starts at ~ 600 psia.
D.	First part is plausible since SIAS has not yet actuated and SIT outlet valves get an open signal when SIAS actuates, however the SIT outlet valves are maintained open when in MODE 1 so they would already be open. Second part is correct.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	7	
Reference Provided:	N	
Learning Objective:	Describe the design characteristics of the Safety Injection Tanks.	

Technical Reference:	Safety Injection System Lesson Plan

EO: 1.16 Describe the design characteristics of the Safety Injection Tanks.

Introduction

The Safety Injection tanks are necessary to mitigate the consequences of a LOCA by providing a rapid source of recovery inventory.

Main Idea

- There are four Safety Injection Tanks provided for the mitigation of a LOCA, one for each cold leg.
- Each SIT contains approximately 14,000 gallons of borated water under approximately 600 psig of nitrogen pressure.
- Each SIT has a motor operated outlet valve that is open and deenergized at NOT/NOP so that a rapid RCS depressurization will result in a discharge of water from the SIT into the RCS without the need for actuation of any supporting system or auxiliary equipment.

The SITs are therefore referred to as a "passive" protection system.

- The SIT outlet valves receive an open signal on a SIAS actuation or if RCS pressure exceeds 410 psia and a permissive to close at 405 psia.
- They are required by TS to be open and deenergized in higher modes.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Reactor Coolant Pump Malfunctions: Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): When to secure RCPs on loss of cooling or seal injection	Tier	1		
	Group	1		
	K/A	015 AA2.10		
	IR	3.7		

Question 4

Per 40AO9-ZZ03, Loss of Cooling Water, what is the MAXIMUM amount of time available to restore cooling flow to the RCPs prior to being procedurally required to trip the Reactor following a complete loss of…

(1) Nuclear Cooling Water ONLY

(2) Nuclear Cooling Water AND Seal Injection flow

A. (1) 10 minutes
(2) 10 minutes

B. (1) 10 minutes
(2) 3 minutes

C. (1) 30 minutes
(2) 10 minutes

D. (1) 30 minutes
(2) 3 minutes

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible since RCPs can be operated indefinitely on a loss of Seal Injection only, so it is reasonable that there would be no additional time constraints on a loss of Seal Injection concurrent with a loss of NC, however when both a lost simultaneously, the limit is 3 minutes.
B.	Correct.
C.	First part is plausible since 30 minutes is the maximum time RCPs can operate without NC before RCP seals will breakthrough, however the procedural limit is 10 minutes. Second part is plausible since RCPs can be operated indefinitely on a loss of Seal Injection only, so it is reasonable that there would be no additional time constraints on a loss of Seal Injection concurrent with a loss of NC, however when both a lost simultaneously, the limit is 3 minutes.
D.	First part is plausible since 30 minutes is the maximum time RCPs can operate without NC before RCP seals will breakthrough, however the procedural limit is 10 minutes. Second part is correct.

Question Source:		New	
	X	Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	3	
Reference Provided:	N	
Learning Objective:	Given the status of NC and RCP seal injection, describe the limitations on RCP operation without NC in accordance with 40AO-9ZZ03.	

Technical Reference:	40AO-9ZZ03, Loss of Cooling Water

4.0 NUCLEAR COOLING WATER

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 1. Enter AOP Entry Time and Date: ________________	
____ 2. IF seal injection is NOT in service, AND cooling water is NOT restored to ANY operating RCP within three minutes of the initial loss, THEN perform the following: a. Ensure the reactor is tripped. b. Stop all of the RCPs. c. Isolate controlled bleedoff.	
____ 3. IF seal injection is in service, AND cooling water is NOT restored to ANY operating RCP within 10 minutes of the initial loss, THEN perform the following: a. Ensure the reactor is tripped. b. Stop all of the RCPs. c. Isolate controlled bleedoff.	

Technical Reference:	LOIT Loss of Cooling Water Lesson Plan

Main Idea

Upon a loss of NC (cooling water) to the RCP(s), operators have thirty (30) minutes to reduce power or isolate cooling water and shutdown the RCP(s). If an RCP is allowed to operate more than 30 minutes without cooling water, possible pump motor assembly bearing seizure may occur.

Explanation

This objective is linked to other lessons.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Reactor Coolant Makeup: Knowledge of the interrelations between the Loss of Reactor Coolant Makeup and the following: Need to avoid plant transients	**Tier**	**1**		
	Group	**1**		
	K/A	**022 AK3.05**		
	IR	**3.2**		

Question 5

Given the following conditions:

- Unit 1 is operating at 100% power
- Charging Pump Mode Selector, CHN-HS-4, is selected to “1-2-3”
- The ‘E’ Charging Pump is aligned to Train ‘B’

In which of the following situations, INDIVIDUALLY, would Prompt and Prudent action be permitted per 40DP-9OP02, Conduct of Operations, in order to prevent a loss of letdown?

1. A loss of Train ‘A’ 4kV Bus, PBA-S03

2. Gas Binding of the ‘A’ Charging Pump

3. Tave1 input to RRS fails LOW

A. 1 ONLY

B. 2 ONLY

C. 1 and 3 ONLY

D. 2 and 3 ONLY

Proposed Answer:	C

Explanations:	
A.	Plausible since this is an allowable condition, however condition 3 is also an allowable condition to use P&P to prevent the loss of letdown
B.	Plausible since the pump to be started is NOT the one which was gas bound, but because the determination of the extent of gas binding (whether it is limited to one pump or if it affects all three pumps) cannot be made without in-field observation, P&P cannot be used in this situation.
C.	Correct.
D.	Plausible since condition 3 does allow for P&P, however condition 2 does not.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	As a licensed operator, perform shift duties and activities in accordance with 40DP-9OP02, Conduct of Operations.	

Technical Reference:	Operations Mentor Guidance

Prompt and Prudent

Prompt and Prudent actions are actions per Conduct of Operations that can be taken with CRS concurrence without first referring to the appropriate procedure. Operations has moved towards the expectation that RO's should announce RK windows by the plate nomenclature and add in RJ information if applicable. The ROs should then move towards the applicable ARP. This is all communicated to CRS. The CRS with the alarm report will determine **if plant needs preclude** the RO reaching the point in the ARP that directs action, and if so, will direct the RO to perform those actions. The ARP guidance should be followed up after the successful performance of the action.

For the purpose of the LOIT 2020 class, **minimal** instances will be allowed to utilize this allowance.

The allowed actions are:
1. Starting the standby stator cooling water pump when the 1st pump trips and 2nd pump fails to start automatically. This is to avoid the 70 second main turbine trip.

2. Starting a second charging pump on obvious instrumentation failures that cause the normally running charging pump to stop. This is to avoid isolating letdown on this failure. This will be accomplished by red-flagging (restarting) the previously running charging pump.

3. Following a Loss of Power to a class bus (LOP), a loss of charging may occur. It is permissible to "green flag" a charging pump that was stopped as a result of the LOP in order to restore charging pump(s) to operation, primarily to prevent letdown from isolating (if it hasn't already) and to restore/maintain charging/seal injection.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Residual Heat Removal System: Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: Service water or closed cooling water pumps	**Tier**	**1**		
	Group	**1**		
	K/A	**025 AK2.03**		
	IR	**2.7**		

Question 6

Given the following conditions:

- Unit 1 is in MODE 4
- SDC is in service on Train ‘B’ using the ‘B’ LPSI Pump

Subsequently:

- The ‘B’ Spray Pond Pump tripped

In order to restore SDC flow using the ‘B’ LPSI Pump, the crew should FIRST attempt to…

A. cross-tie Plant Cooling Water to ‘B’ Spray Pond Cooling Water to restore cooling to the ‘B’ EW Heat Exchanger

B. cross-tie ‘B’ Nuclear Cooling Water to ‘B’ Essential Cooling Water to restore cooling to the ‘B’ SDC Heat Exchanger

C. start and align the ‘A’ Spray Pond Pump to the ‘B’ EW Heat Exchanger to restore cooling to the ‘B’ EW Heat Exchanger

D. place Train ‘A’ Spray Pond / Essential Cooling / Essential Chill Water in service and align the ‘B’ LPSI Pump to the ‘A’ SDC Heat Exchanger

Proposed Answer:	D

Explanations: **In order to prevent cueing the correct answer to the second part of question 30, I intentionally did not reference the procedure which directs this action. There is only one correct way to respond to this event procedurally so I don't believe it is necessary to include the "Per xxx…" intro to this question.**	
A.	Plausible since non-class cooling systems can be cross-tied with class cooling systems to restore cooling flow (NC to EW), however Plant Cooling cannot be aligned to supply Spray Pond Cooling
B.	Plausible as NC can be used to supply EW flow, however this is only an option is both trains of class auxiliaries are not available.
C.	Plausible that either Spray Pond Pump can be used to supply cooling flow to the 'B' EW heat exchanger since either NC pump can be used to supply EW flow, however Spray Pond is train dependent.
D.	Correct.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

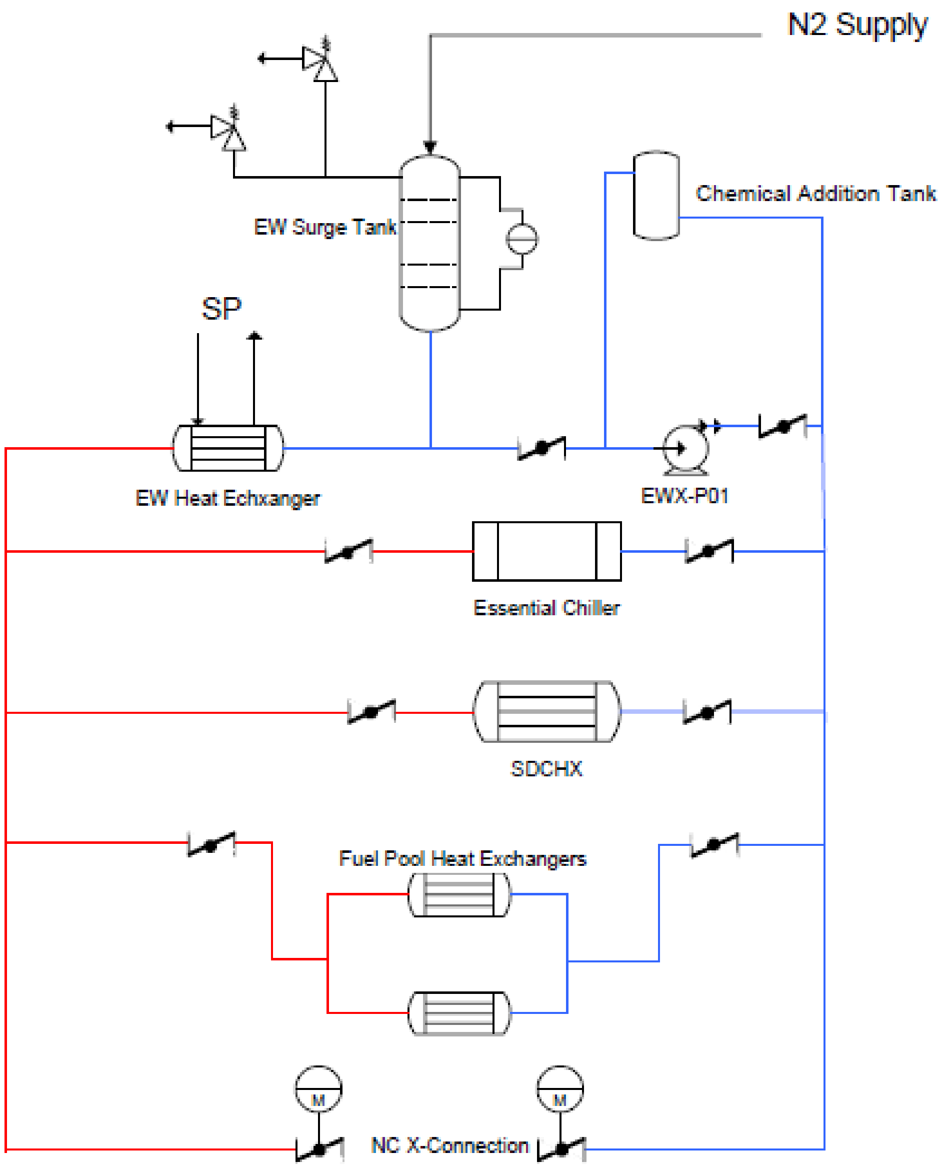
Level of Difficultly:	3	
10CFR55.41:	8	
Reference Provided:	N	
Learning Objective:	Given that the LMFRP is being performed and HR is in progress, outline the major steps used to control Core and RCS Heat Removal in the HR success paths per 40EP-9EO11.	

Technical Reference:	40EP-9EO11, Lower Mode Functional Recovery

INSTRUCTIONS	CONTINGENCY ACTIONS
* 10. IF the SDC train auxiliaries are NOT available, THEN perform the following: a. IF the SDC auxiliaries are NOT available to the appropriate train, THEN PERFORM ONE of the following: • Appendix 241, LM - SDC Train A using Train B Auxiliaries • Appendix 242, LM - SDC Train B using Train A Auxiliaries b. IF BOTH of the following: • Both SDC train auxiliaries are NOT available • The Nuclear Cooling Water System is available THEN PERFORM ONE of the following: • Appendix 243, LM - NC Cross Tie to EW Train A • Appendix 244, LM - NC Cross Tie to EW Train B	

Technical Reference:	Essential Cooling Water Tech Manual

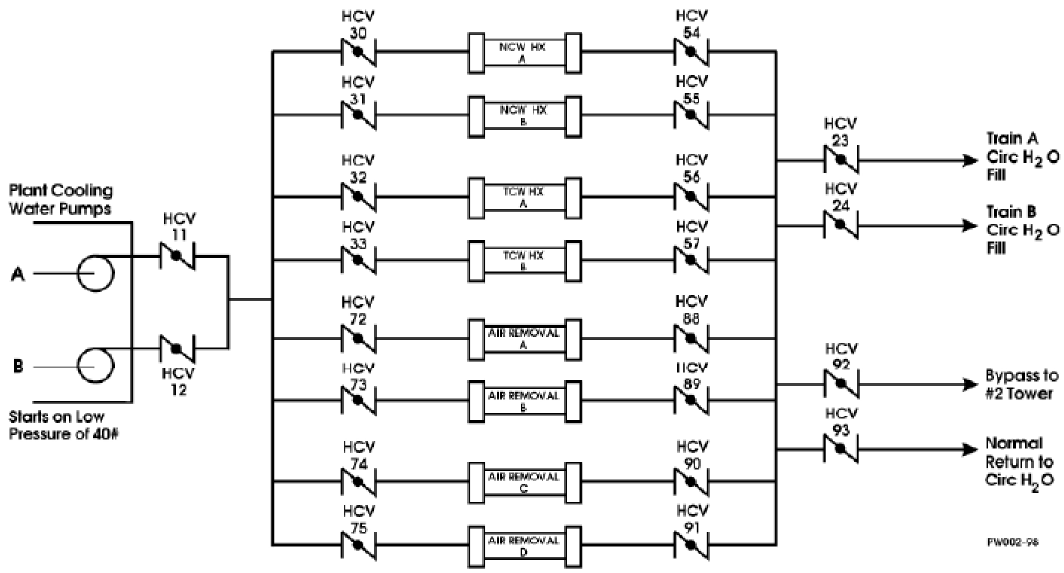
The spray pond system cools the EWHX which in turn cools the SDC HX, however the A and B trains of SP cannot be cross-tied to cool opposite train EW systems.



Technical Reference:	Plant Cooling Water System Tech Manual

Although the Plant Cooling Water System does cool the NC system, and is the non-class version of the spray pond system, it cannot be used to provide cooling for SDC following the loss of the Spray Pond pump

PLANT COOLING WATER SYSTEM



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Component Cooling Water: Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: Effect on the CCW flow header of a loss of CCW	Tier	1		
	Group	1		
	K/A	026 AK3.04		
	IR	3.5		

Question 7

Which of the following reasons explain why Train ‘A’ EW is used instead of Train ‘B’ when cross-tying NC and EW following a loss of NC while at power?

1. Because Train ‘A’ EW takes less time to cross-tie to NC than Train ‘B’ EW
2. To ensure cooling will be maintained to the RCPs in the event of a SIAS
3. Because Train ‘A’ EW cross-tie valves will auto close on an ‘A’ EW Surge Tank Low Level signal

A. 1 and 2 ONLY

B. 1 and 3 ONLY

C. 2 ONLY

D. 3 ONLY

Proposed Answer:	B

Explanations:	
A.	1 is correct. 2 is plausible since it is highly desired to maintain forced circ during a SGTR, which would result in a SIAS, however if Train ‘A’ EW is cross-tied with NC, RCPs will have to be stopped if SIAS occurs since the cross-tie valves will auto close.
B.	Correct.
C.	2 is plausible since it is highly desired to maintain forced circ during a SGTR, which would result in a SIAS, however if Train ‘A’ EW is cross-tied with NC, RCPs will have to be stopped if SIAS occurs since the cross-tie valves will auto close. Plausible that 1 would not be true since almost all systems with two trains of equipment in the control room would take the same amount of time to align, however Train ‘B’ EW cross-tie valves are manually actuated valves which can only be operated in the field.
D.	3 is correct, however 1 is also correct. Plausible that 1 would not be true since almost all systems with two trains of equipment in the control room would take the same amount of time to align, however Train ‘B’ EW cross-tie valves are manually actuated valves which can only be operated in the field.

Question Source:	x	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	4	
Reference Provided:	N	
Learning Objective:	From memory, describe the interlocks associated with the Train 'A' EW to NC cross tie valves (EWA-UV-145 and EWA-UV-65)	

Technical Reference:	LOIT Loss of Cooling Water Lesson Plan

Because the crew has a limited amount of time to complete the cross-tie and restore cooling to the RCPs, Train A is generally used instead of Train B due to the cross-tie valves being able to be operated from the control room (MOVs) as opposed to Train B cross-tie valves which are large manual valves which must be operated in the field. Because the Train B valves are manually operated, they do not close on a SIAS actuation (which would maintain cooling to the RCPs following a SIAS), however that is not a desired condition if SIAS actutates since the cooling flow is needed for more important loads. Additionally, the Train A cross-tie valves close on a low level signal, automatically preventing the complete loss of a train of essential cooling water due to a leak.

UFSAR 18.II.K.3.25

The reactor coolant pump normal cooling water system (nuclear cooling water system (NCWS) is backed up by the essential cooling water system (ECWS) to supply cooling water to the seals during a loss of offsite power. In the event of a loss of offsite power, the operator can open the train A-NCWS crosstie valves from the control room, permitting the ECWS train A to supply cooling water to the reactor coolant pump seals. If train A fails, the operator must manually open the train B-NCWS crosstie valves and shut the train A crosstie valves to permit the same function. The crosstie of the ECWS to supply the NCWS priority heat loads is described in a PVNGS Station Manual procedure which allows 10 minutes for the operator to align the ECWS.

Main Idea

If cooling water is lost to the RCPs due to a LOOP, the operator has ten (10) minutes to supply cooling water to the RCP seals from the essential cooling water system.

If this time is not met and seal injection is in service, 40AO-9ZZ03 directs the operator to trip the Reactor, stop all RCPs, isolate controlled bleedoff, and perform the appropriate procedure for plan conditions. Also, operation of the RCPs without cooling water may result in damage to pump thrust bearings and possible bearing seizure.

Explanation

This objective is linked to other lessons.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Pressurizer Pressure Control System Malfunction: Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners	Tier	1		
	Group	1		
	K/A	027 AK2.03		
	IR	2.6		

Question 8

Given the following conditions:

- Unit 3 is operating at 100% power
- RCS Pressure is stable at the current setpoint of 2250 psia
- All Pressurizer Backup Heaters are OFF
- Both Pressurizer Proportional Heaters are ON
- RCN-HS-100, Pressure Control Channel X/Y Selector, is selected to Channel 'Y'

Subsequently:

- RCN-PT-100Y, Pressurizer Control Channel 'Y', failed to 1500 psia

With NO operator action, RCN-PIC-100, Pressurizer Pressure Control, OUTPUT will go from an INITIAL value of approximately ___(1)___ to a FINAL value of ___(2)___ .

A. (1) 16.5%
(2) 0%

B. (1) 16.5%
(2) 100%

C. (1) 33%
(2) 0%

D. (1) 33%
(2) 100%

Proposed Answer:	A

Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible since proportional heater output will go to 100%, however PIC-100 is reverse acting, so PIC-100 output goes to 0%.
C.	First part is plausible since at NOP, PIC-100 output is ~ 1/3 of the useable range, however even though controller output does go from 0-100%, the useable range is 0-50%, making the normal operating output of PIC-100 ~16.5%. Second part is correct.
D.	First part is plausible since at NOP, PIC-100 output is ~ 1/3 of the useable range, however even though controller output does go from 0-100%, the useable range is 0-50%, making the normal operating output of PIC-100 ~16.5%. Second part is plausible since proportional heater output will go to 100%, however PIC-100 is reverse acting, so PIC-100 output goes to 0%.

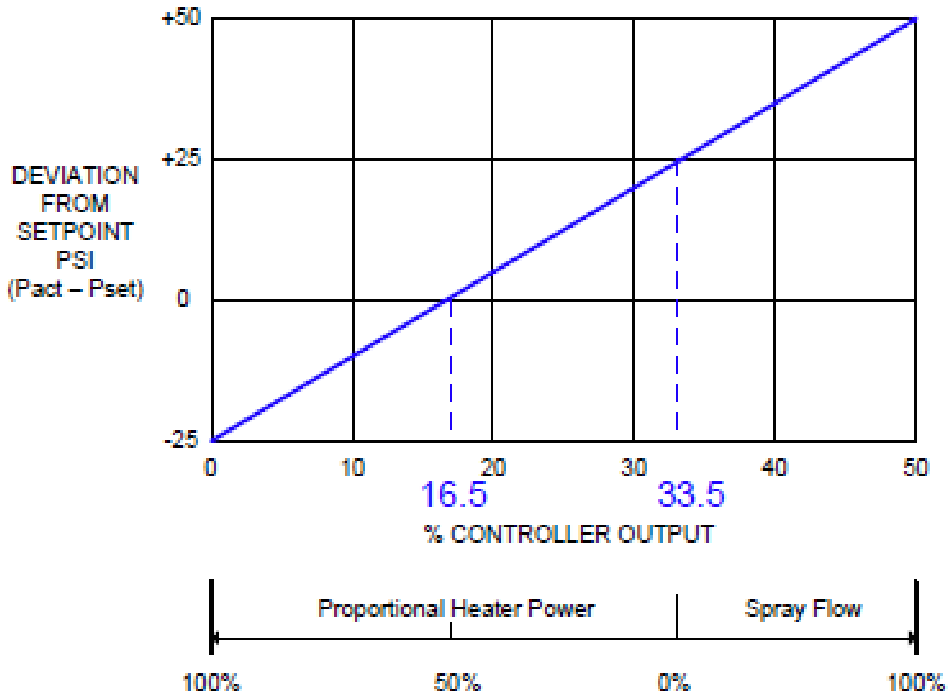
Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	3	
Reference Provided:	N	
Learning Objective:	Describe the manual/automatic functions associated with the Pressurizer Pressure Control System.	

Technical Reference:	PPCS System Tech Manual

Pressurizer Pressure Control System (PPCS)



Master Controller Output

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Anticipated Transient Without Scram: Ability to determine or interpret the following as they apply to a ATWS: System component valve position indications	**Tier**	**1**		
	Group	**1**		
	K/A	**029 EA2.05**		
	IR	**3.4**		

Question 9

Given the following conditions:

- Unit 3 is operating at 100% power
- RPCB is OOS for corrective maintenance

Subsequently:

- The Main Turbine tripped
- All RPS trips failed to trip the Reactor
- The Reactor automatically tripped via the Supplemental Protection System

10 seconds after the Supplemental Protection System actuates, assuming NO operator actions have been taken, the Pressurizer Safety Valves will be ___(1)___ and the MSIVs will be ___(2)___ .

A. (1) open
(2) open

B. (1) open
(2) closed

C. (1) closed
(2) open

D. (1) closed
(2) closed

Proposed Answer:	C

Explanations:	
A.	First part is plausible since the SPS actuation and the lifting of PSVs are both triggered on high RCS pressure, however the SPS actuates at 2409 psia and the PSVs don't lift until 2450 psia. Second part is correct.
B.	First part is plausible since the SPS actuation and the lifting of PSVs are both triggered on high RCS pressure, however the SPS actuates at 2409 psia and the PSVs don't lift until 2450 psia. Second part is plausible since power follows steam demand and the closure of MSIVs on receipt of an SPS trip would limit the positive reactivity addition from the ongoing steam demand from steam driven components, however the SPS does not send a close signal to the MSIVs.
C.	Correct.
D.	First part is correct. Second part is plausible since power follows steam demand and the closure of MSIVs on receipt of an SPS trip would limit the positive reactivity addition from the ongoing steam demand from steam driven components, however the SPS does not send a close signal to the MSIVs.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	6	
Reference Provided:	N	
Learning Objective:	Describe the Supplementary Protection System including its function, instrumentation, bases, and setpoint	

Technical Reference:	Plant Protection Tech Manual

SPS setpoint (ATWS trip high pressure) is 2409 psia

PARAMETER MONITORED	INST #	FUNCTION					RANGE/ SETPOINTS	FUNCTION
		LOC	CR	ALARM	COMP	CONT		
Low PZR Pressure	Bistable 6 PPS CH D		X	X	X	X	Pre-trip 1880 psia Trip 1837 psia	Reactor Trip/SIAS/CIAS
PZR Pressure	RCA-PT-0199A	X					1500 - 2500 psia	SPS PZR Pressure Transmitter
PZR Pressure	RCB-PT-0199B	X					1500 - 2500 psia	SPS PZR Pressure Transmitter
PZR Pressure	RCC-PT-0199C	X					1500 - 2500 psia	SPS PZR Pressure Transmitter
PZR Pressure	RCD-PT-0199D	X					1500 - 2500 psia	SPS PZR Pressure Transmitter
Hi PZR Pressure	SPS CH A		X	X	X	X	2409 PSIA	Reactor Trip
Hi PZR Pressure	SPS CH B		X	X	X	X	2409 PSIA	Reactor Trip
Hi PZR Pressure	SPS CH C		X	X	X	X	2409 PSIA	Reactor Trip
Hi PZR Pressure	SPS CH D		X	X	X	X	2409 PSIA	Reactor Trip
SG #1 Pressure	SGA-PT-1013A	X					0 - 1524 psia	PPS SG #1 Pressure

Technical Reference:	RCS Tech Manual

The pressurizer safety valves are set to open at 2475 +/- 25 psia, therefore would open at 2450 psia at the earliest, which is ~ 40 psia higher than the SPS trip would occur, therefore the safeties would NOT be open.

Figure 2 - 38 Pressurizer

Primary Safety Valves (PSV-200, 201, 202, 203)

The function of the safety valves (PSV-200, 201, 202 and 203) is to limit the RCS pressure to less than the RCS safety limit pressure of 2750 psia. The pressurizer is equipped with four safety valves. Each safety valve is on a separate line connected to the top of the pressurizer. The safety valves are totally enclosed, backpressure compensated, spring loaded, self-activated, pop-type valves. The valves are set to open at 2475 psia, ± 25 psia with a 3% accumulation. The blowdown factor is 5%. The combined

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Steam Generator Tube Rupture: Ability to operate and monitor the following as they apply to a SGTR: S/G sample isolation valve indicators	**Tier**	**1**		
	Group	**1**		
	K/A	**038 EA1.17**		
	IR	**3.2**		

Question 10

Given the following conditions:

- Unit 1 was manually tripped due to a SGTR on SG #1
- The crew manually actuated SIAS and CIAS following the trip

The SG Sample Valves will close due to the ___(1)___ actuation, and following entry into 40EP-9EO04, SGTR, the SG Sample Valves will be overridden and opened on ___(2)___ .

A. (1) SIAS
(2) SG #1 ONLY

B. (1) SIAS
(2) SG #1 AND SG #2

C. (1) CIAS
(2) SG #1 ONLY

D. (1) CIAS
(2) SG #1 AND SG #2

Proposed Answer:	**B**

Explanations:	
A.	First part is correct. Second part is plausible since SG #1 is the only SG with a tube rupture, however following entry into the SGTR EOP, both SG sample valves are overridden and opened.
B.	Correct.
C.	First part is plausible since the CIAS does isolate sample lines, however it isolates RCS sample lines, not SG sample lines. Second part is plausible since SG #1 is the only SG with a tube rupture, however following entry into the SGTR EOP, both SG sample valves are overridden and opened.
D.	First part is plausible since the CIAS does isolate sample lines, however it isolates RCS sample lines, not SG sample lines. Second part is correct.

Question Source:	**x**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		**Memory or Fundamental Knowledge**
	x	**Comprehension or Analysis**

Level of Difficultly:	**3**	
10CFR55.41:	**10**	
Reference Provided:	**N**	
Learning Objective:	**Given that the SGTR EOP is being implemented, describe the SGTR EOP mitigation strategy in accordance with 40EP-9EO04.**	

Technical Reference:	Main Steam Tech Manual

2.2.10 Downcomer Blowdown Sample Containment Isolation Valves

Downcomer blowdown sample containment isolation valves (see figure 2-17).

Upstream, inside containment isolation valves:

- SGA-UV-220 (SG-1)
- SGB-UV-226 (SG-2)

Downstream, outside containment isolation valves:

- SGB-UV-221 (SG-1)
- SGA-UV-227 (SG-2)

The blowdown sample containment isolation valves are solenoid operated, normally closed, 1/2" globe valves.

Sample Valve Controls

One, three position (OPEN/CLOSE), spring return-to-normal control switch is provided in the control room for each of the blowdown sample isolation valves. The blowdown sample isolation valves all fail closed on loss of power and close upon receipt of AFAS-1, AFAS-2, MSIS, or SIAS. Following automatic closure, the control room operator can override the auto close signal and open the valves by momentarily placing the control switch in CLOSE, and then in OPEN. When moved to CLOSE, a white OVERRIDE light illuminates, indicating the valve can be overridden.

2.2.11 MSIV Bypass Valves (UV-169, UV-183)

The main steam isolation valve bypass valves are 4 inch electro-pneumatic gate valves. (figure 2-18.)

4.5 Instructions/Contingency Actions

4.5.1 Step 1 - Monitor the SFSCs

A. This step directs actions that will ensure that the correct procedure is implemented for the event in progress.

The diagnosis of a SGTR is confirmed by meeting all acceptance criteria in the Safety Function Status Check. This action ensures that the proper procedure is being used to mitigate the event. In particular, the CRS should note the status of RCS subcooling, containment radiation level and steam plant activity. These parameters provide a means of discriminating between SGTRs, LOCAs and ESDs.

- For a SGTR, steam plant activity monitors may be alarming, but containment activity monitors should not be alarming.
- For a LOCA, the RCS may reach saturated conditions and containment activity monitors may be alarming but steam plant activity monitors should not be alarming.
- For an ESD, neither steam plant nor containment activity monitors should be alarming. For units which exhibit SG tube leakage, however, steam plant or containment activity monitors may alarm during ESDs.

LOCAs, ESDs, and SGTRs have similar initial symptoms and could be confused early in the event. Sampling for SGTR will ensure that the appropriate samples are drawn, including sampling both SGs for activity, which will assist the CRS in confirming the diagnosis of a SGTR.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Steam Line Rupture-Excessive Heat Transfer: Ability to operate and / or monitor the following as they apply to the (Excess Steam Demand): Desired operating results during abnormal and emergency situations	**Tier**	**1**		
	Group	**1**		
	K/A	**CE E05 EA1.3**		
	IR	**3.4**		

Question 11

Given the following conditions:

- Unit 1 was tripped due to an ESD outside of Containment
- MSIS was manually actuated
- All Thot and Tcold indications are lowering

The crew should commence depressurizing the intact SG as soon as the most affected SG ___(1)___ and the intact SG should be stabilized at saturation pressure for the lowest observed RCS ___(2)___ in the loop of the most affected SG.

A. (1) is identified
   (2) Thot

B. (1) is identified
   (2) Tcold

C. (1) reaches rebound
   (2) Thot

D. (1) reaches rebound
   (2) Tcold

Proposed Answer:	D

Explanations:	
A.	First part is plausible since the sooner the intact SG is depressurized, the less impactful the repressurization of the RCS will be following dryout of the affected SG, however at PNVGS, depressurization of the intact SG is not commenced until the affected SG reaches rebound (dryout). Second part is plausible since stabilizing to the lowest RCS Thot would stabilize RCS temperature and would minimize the amount of the additional RCS cooldown, however Tcold is used for stabilization of RCS temperature following an ESD at PVNGS.
B.	First part is plausible since the sooner the intact SG is depressurized, the less impactful the repressurization of the RCS will be following dryout of the affected SG, however at PNVGS, depressurization of the intact SG is not commenced until the affected SG reaches rebound (dryout). Second part is correct.
C.	First part is correct. Second part is plausible since stabilizing to the lowest RCS Thot would stabilize RCS temperature and would minimize the amount of the additional RCS cooldown, however Tcold is used for stabilization of RCS temperature following an ESD at PVNGS.
D.	Correct.

Question Source:	x	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	**Memory or Fundamental Knowledge**
		Comprehension or Analysis

Level of Difficultly:	**3**	
10CFR55.41:	**10**	
Reference Provided:	**N**	
Learning Objective:	**Given a set of plant parameters, determine when and how RCS temperature is stabilized during an ESD per 40EP-9EO05.**	

ESD Step 14

1. This step minimizes the effects of the temperature rebound which occurs when the faulted SG dries out. The basis states that a controlled heat removal method should be established before the dryout condition occurs. This does not mean to follow the faulted SG pressure down with the good SG. Doing so invalidates the safety analysis of the UFSAR. This statement is only intended to ensure a heat removal method is available on the good SG. Lowering the good SG pressure to the established target shall only be done when the faulted SG is dry. The target pressure for the non-faulted SG is determined from the lowest Tc of the RCS. When the faulted SG dries out, as indicated by RCS temperatures rising, the lowest Tc is noted and the saturation pressure for this temperature is determined from the steam tables. This pressure is the target pressure for the non-faulted SG. If the operator recognizes Tc has risen sometime after SG dry out, they should use the observed Tc at that time to determine the target pressure for the intact SG in order to stabilize at the existing plant conditions and prevent another unnecessary cooldown.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Main Feedwater: Knowledge of the operational implications of EOP warnings, cautions, and notes	Tier	1		
	Group	1		
	K/A	054 G 2.4.20		
	IR	3.8		

Question 12

Given the following conditions:

- Unit 1 tripped from 100% power due to a loss of both Main Feedwater Pumps
- The CRS has directed the BOP to perform Appendix 44, Feeding With the Condensate Pumps, to restore feedwater
- Both SG levels are 10% WR
- Both SG pressures are 1170 psia
- SG #1 has been selected for the restoration of feedwater

(1) Per 40DP-9AP17, Standard Appendices Technical Guideline, during the depressurization of SG #1, the 100°F/hr cooldown rate…

(2) If SG #1 WR level reaches 0% WR prior to SG #1 pressure lowering to less than Condensate Pump discharge pressure, the BOP should…

A. (1) MAY be exceeded
(2) continue depressurizing SG #1 until pressure is less than Condensate Pump discharge pressure

B. (1) MAY be exceeded
(2) stop depressurizing SG #1 and commence depressurizing SG #2 to establish Condensate feed on SG #2

C. (1) may NOT be exceeded
(2) continue depressurizing SG #1 until pressure is less than Condensate Pump discharge pressure

D. (1) may NOT be exceeded
(2) stop depressurizing SG #1 and commence depressurizing SG #2 to establish Condensate feed on SG #2

Proposed Answer:	A

Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible since the technical guidelines state that if one SG is dry and the other contains water, that feedwater should not be added to the dry SG and rather the SG with inventory should be fed, however simply lowering below 0% WR does not mean the SG is dry and lowering from 10% WR to < 0% WR is likely during a rapid depressurization in order to feed with Condensate Pumps.
C.	First part is plausible as 100°F/hr is the cooldown rate limit at PVNGS and is not generally allowed to be exceeded, however since core heat removal is lost, restoring this capability promptly takes precedence over maintaining the 100°F/hr cooldown rate. Second part is correct.
D.	First part is plausible as 100°F/hr is the cooldown rate limit at PVNGS and is not generally allowed to be exceeded, however since core heat removal is lost, restoring this capability promptly takes precedence over maintaining the 100°F/hr cooldown rate. Second part is plausible since the technical guidelines state that if one SG is dry and the other contains water, that feedwater should not be added to the dry SG and rather the SG with inventory should be fed, however simply lowering below 0% WR does not mean the SG is dry and lowering from 10% WR to < 0% WR is likely during a rapid depressurization in order to feed with Condensate Pumps.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		**Memory or Fundamental Knowledge**
	X	**Comprehension or Analysis**

Level of Difficultly:	**3**	
10CFR55.41:	**10**	
Reference Provided:	**N**	
Learning Objective:	**Given conditions of a LOAF and the status of plant equipment, determine from where feed can be established per 40EP-9EO06.**	

Technical Reference:	Appendix 44, Feeding With the Condensate Pumps

Once a SG is selected for use, that SG is depressurized until it is being fed unless conditions cannot be established. Even if level drops below 0% WR, the SG is not "dry" as there is ~ 30 minutes of steaming inventory below 0% WR. Additionally, even if the selected SG does go dry, there are still allowances to feed the dry SG, albeit at a reduced feed rate.

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 14. **IF** Steam Generator #1 was selected, **THEN** perform the following:	
a. Fast close Steam Generator #1 MSIVs by using the following pushbuttons: • SGA-HS-251 • SGB-HS-253	
b. Lower Steam Generator #1 pressure below the condensate pump discharge pressure using SG 1 ADVs.	b.1 PERFORM Appendix 18, Local ADV Operation.
c. Maintain Steam Generator #2 pressure less than 1200 psia.	
d. **IF** Steam Generator #1 is dry, **THEN** maintain feed flow rate of less than or equal to 1000 gpm (0.5×10^6 lbm/hr).	
e. **IF** using SG 1 Downcomer Control valve, **THEN** throttle open SGN-FV-1113.	
f. **IF** using SG 1 Downcomer Bypass valve, **THEN** throttle open SGN-HV-1143.	
g. GO TO Step 16.	

Technical Reference:	40DP-9AP17, Standard Appendices Technical Guideline

4.1.44 Appendix 44 - Feeding with the Condensate Pumps

A. This appendix will provide guidance to align a Condensate Pump to supply the SG. This procedure assumes the normal flow path for supplying the SG is still available. Operator should evaluate each Steam Generator to determine which steam generator could successfully provide heat removal capabilities (able to be fed and steamed).

- SG Press - SG with lowest pressure will take less time to depressurize.
- SG Level - SG with lowest level will take less time to depressurize.
- Ability to be fed from the condensate system - Unit can successfully accomplish a lineup that would provide condensate flow to the SG and the SG can be steamed.

This appendix aligns a flow path to the downcomer region of the SG. If a loss of M41 (SGA-UV-172/175) or M42 (SGB-UV-130/135) had occurred the operator may have to verify the position of the Downcomer Isolations locally or by observing indicated feed flow once the pump is started and the Downcomer Control Valve is opened. Placing the Downcomer Control Valve(s) in manual will provide the operator with the ability to control the feed flow to the steam generators. The cooldown should be performed using the atmospheric dump valves (ADVs), this gives the operator the ability to de-pressurize one of the SGs and conserve inventory in the non-selected SG. The maximum allowed cooldown rate of 100°F/hr may be exceeded during steam generator depressurization and subsequent refill with condensate. Reestablishing feedwater to recover heat removal capabilities has priority over the consequences of over cooling. PTS should be a concern anytime the RCS has undergone a rapid cooldown and depressurization, care should be taken to not allow the RCS to heat up or re-pressurize.

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Loss of Offsite Power: Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: Principle of cooling by natural convection	**Tier**	**1**	[redacted]	
	Group	**1**	[redacted]	
	K/A	**056 AK1.01**		
	IR	**3.7**	[redacted]	

Question 13

Given the following conditions:

- Unit 1 tripped from 100% power due to a loss of offsite power
- The crew is verifying that natural circulation has been established

As natural circulation flow develops, the crew should expect to see Loop ΔT indicating ___(1)___ 65°F and should expect a delay of approximately ___(2)___ minutes before the RCS temperature response to feeding and steaming adjustments can be verified.

A. (1) LESS than
(2) 1 to 2

B. (1) LESS than
(2) 5 to 15

C. (1) GREATER than
(2) 1 to 2

D. (1) GREATER than
(2) 5 to 15

Proposed Answer:	B

Explanations:	
A.	First part is correct. First part is correct. Second part is plausible since frequent adjustments of steaming and feeding are needed when controlling in manual (as is the case in a LOOP/LOFC) in order to maintain parameters within post-trip control bands, however in natural circulation conditions, the plant response to these adjustments will not be seen for ~ 5 to 15 minutes.
B.	Correct.
C.	First part is plausible since the driving head in natural circulation is developed by the difference in density between the hot and cold legs, therefore a higher delta-T than with forced circulation is plausible, however delta-T must be < 65°F (full power delta-T) in natural circulation conditions. Second part is plausible since frequent adjustments of steaming and feeding are needed when controlling in manual (as is the case in a LOOP/LOFC) in order to maintain parameters within post-trip control bands, however in natural circulation conditions, the plant response to these adjustments will not be seen for ~ 5 to 15 minutes.
D.	First part is plausible since the driving head in natural circulation is developed by the difference in density between the hot and cold legs, therefore a higher delta-T than with forced circulation is plausible, however delta-T must be < 65°F (full power delta-T) in natural circulation conditions. Second part is correct.

Question Source:		New	
	X	Bank	
		Modified	
	X	Previous NRC Exam	2020 RO Exam Q57

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	14	
Reference Provided:	N	
Learning Objective:	Given a loss of forced circulation, identify the parameters used to determine Natural Circulation flow per 40EP-9EO07.	

Technical Reference:	40DP-9AP13, Blackout Technical Guideline

4.5.22 Step 22 - Ensure Natural Circulation

A. The intent of this step is to check that natural circulation flow is established and is supporting RCS heat removal. After RCPs are tripped, natural circulation flow should develop within 5 - 15 minutes (longer if the plant tripped from a low power). Natural circulation flow will continue as long as RCS pressure and inventory control are maintained and at least one steam generator is available for heat removal. Natural circulation flow is determined by a combination of factors. The factors which affect natural circulation include decay heat, component elevations, primary to secondary heat transfer, loop flow restrictions, and voiding. The component elevations are such that satisfactory natural circulation decay heat removal is obtained utilizing density differences between the bottom of the core and the top of the steam generator tube sheet. These density differences occur when primary to secondary heat removal through the steam generator tubes is utilized. When single phase natural circulation flow is established in at least one loop, the RCS should indicate the following conditions:

- Loop Delta-T less than normal full power Delta-T. This ensures by plant design that the Power/Flow ratio remains less than 1. A Power/Flow ratio of less than 1 ensures that heat can be removed from the RCS during the establishment of natural circulation. Initially T_h may increase causing the loop Delta-T to rise but once natural circulation is established the loop Delta-T will drop.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Vital AC Instrument Bus: Knowledge of annunciator alarms, indications, or response procedures	**Tier**	**1**		
	Group	**1**		
	K/A	**057 G 2.4.31**		
	IR	**4.2**		

Question 14

Given the following conditions:

- Unit 3 is operating at 100% power
- RCN-HS-110, Level Control Selector switch, is selected to ‘X’
- RCN-HS-100, Pressure Control Selector switch, is selected to ‘X’
- RCN-HS-100-3, Heater Control Selector switch, is selected to ‘BOTH’

Subsequently:

- A fault caused a loss of Train ‘A’ Class Instrument Bus, PNA-D25

Per 40AO-9ZZ13, Loss of Class Instrument or Control Power, which of the following switches must be placed in Channel ‘Y’?

1. RCN-HS-110, Level Control Selector switch
2. RCN-HS-100, Pressure Control Selector switch
3. RCN-HS-100-3, Heater Control Selector switch

A. 1 ONLY

B. 2 ONLY

C. 1 and 3 ONLY

D. 2 and 3 ONLY

Proposed Answer:	C

Explanations:	
A.	Plausible since RCN-HS-100 must be placed in Channel Y, however HS-100-3 must also be placed in Y.
B.	Plausible since placing HS-100 to Channel Y would be required if the X pressure input was lost, however this input is non-class and is therefore unaffected. Also plausible that HS-110 and 100-3 would be unaffected since some instrument power sources are “counter-intuitively” aligned – for example non-class instrument buses NNN-D11 and NNN-D12. NNN-D11 powers the Channel Y pressurizer pressure input and NNN-D12 powers the Channel X input.
C.	Correct.
D.	Plausible since 100-3 is correct, however the level selector is also affected, not the pressure selector.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	7	
Reference Provided:	N	
Learning Objective:	Given a loss of PKA or PKB, describe how PZR pressure is controlled in accordance with 40AO-9ZZ13.	

Technical Reference:	40AO-9ZZ13, Loss of Class Instrument or Control Power

4.0 LOSS OF PNA-D25

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 1. Enter AOP Entry Time and Date: ________________	
____ 2. Ensure the event is being classified.	
___ 3. Ensure **BOTH** of the following handswitches are selected to Channel Y: • RCN-HS-100-3, Heater Control Selector switch • RCN-HS-110, Level Control Selector switch	

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Loss of DC Power: Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation	Tier	1	[redacted]	
	Group	1	[redacted]	
	K/A	058 AK1.01		
	IR	2.8	[redacted]	

Question 15

Given the following conditions:

- Unit 1 is operating at 100% power
- 125 VDC Bus, PKC-M43, is being powered from the ‘C’ Battery Charger
- 125 VDC Bus, PKA-M41, is being powered from the ‘A’ Battery Charger
- 120 VAC Bus, PNC-D27, is being powered from Inverter N13

Subsequently:

- The DC Output Breaker from the ‘C’ Battery Charger to PKC-M43, was inadvertently opened by an instructor conducting a JPM

With NO operator action, 125 VDC Bus, PKC-M43, will be powered from ___(1)___ , and 120 VAC Bus, PNC-D27, will be powered from ___(2)___ .

A. (1) the ‘C’ Battery
   (2) Inverter PNC-N13

B. (1) the ‘C’ Battery
   (2) Voltage Regulator PNC-V27

C. (1) the ‘AC’ Swing Battery Charger
   (2) Inverter PNC-N13

D. (1) the ‘AC’ Swing Battery Charger
   (2) Voltage Regulator PNC-V27

Proposed Answer:	A

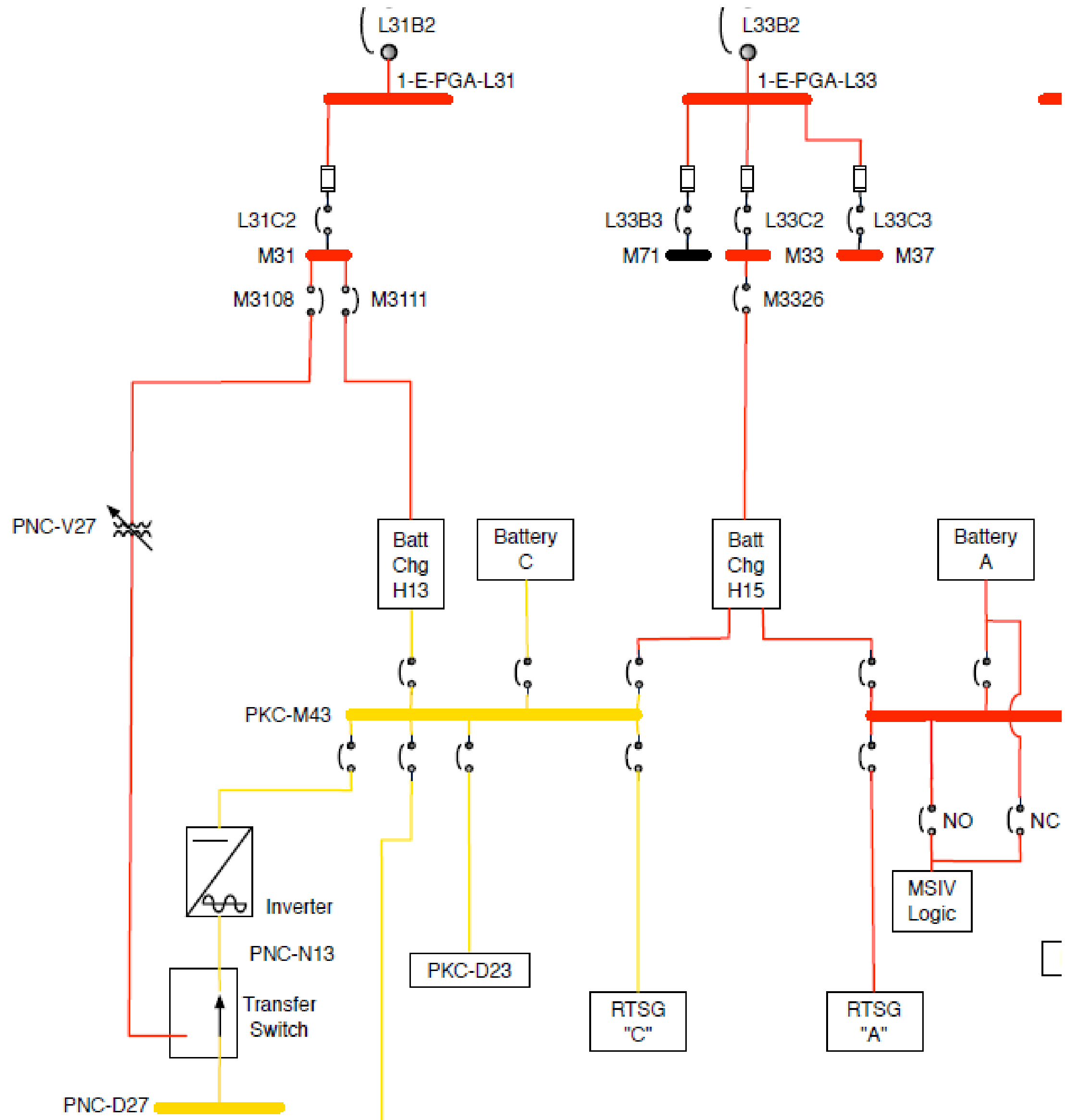
Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible as this would be case if power was lost to the 125 VDC bus PKC, however since power is maintained to the DC bus, PNC remains aligned to the inverter.
C.	First part is plausible since the AC charger can be aligned to the PKC bus, and this is a more desired alignment than having the bus being powered from the battery, however alignment of the swing charger is a manual alignment and does not happen automatically. Second part is correct.
D.	First part is plausible since the AC charger can be aligned to the PKC bus, and this is a more desired alignment than having the bus being powered from the battery, however alignment of the swing charger is a manual alignment and does not happen automatically. Second part is plausible as this would be case if power was lost to the 125 VDC bus PKC, however since power is maintained to the DC bus, PNC remains aligned to the inverter.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	7	
Reference Provided:	N	
Learning Objective:	Explain the operation of the Class 1E 125 VDC Batteries under normal operating conditions.	

Battery Charger H15 is the AC Swing Charger. If the output of the C charger (H13) is lost, the C battery will automatically power PKC. The inverter will remain aligned since voltage was never lost to PKC.



Technical Reference:	LOIT 120 VAC Power Lesson Plan

Main Idea

The PN System consists of four independent ungrounded subsystems (Channels A, B, C and D) each containing a DC to AC inverter, a backup AC voltage regulator (i.e., a regulating step down transformer), a static (automatic) transfer switch, a distribution panel and associated connected loads. The inverters are fed from the Class 1E 125V DC Power (PK) System MCCs while the voltage regulators are fed from the Class 1E 480V AC Power (PH) System MCCs. Each of the four subsystems is dedicated to and provides 120V AC instrumentation and control power to one of the four redundant and independent channels of the Reactor Protection (SB) System and the Engineered Safety Features Actuation (SA) System.

Under normal operation of the system, the inverters receive 125V DC power from the PK System. As an alternate (standby or bypass) source of power for operation of the system on loss of an inverter, the voltage regulators receive 480V AC power from the PH System and provide 120V AC, single phase, ungrounded, 60 Hz power (via transfer switches) to the distribution panels and their connected loads.

Transfer from normal to alternate power operation on loss of an inverter is done automatically for all three units. On return of the inverter (normal) power, manual operation is required.

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Loss of Nuclear Service Water: Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS	**Tier**	**1**	[redacted]	
	Group	**1**	[redacted]	
	K/A	**062 AK3.02**		
	IR	**3.6**	[redacted]	

Question 16

Nuclear Cooling Water Containment Isolation Valves receive an automatic close signal on a ___(1)___ actuation in order to mitigate the effects of ___(2)___ .

A. (1) CIAS
   (2) a high energy release inside Containment

B. (1) CIAS
   (2) an RCS to NC intersystem Loss of Coolant Accident

C. (1) CSAS
   (2) a high energy release inside Containment

D. (1) CSAS
   (2) an RCS to NC intersystem Loss of Coolant Accident

Proposed Answer:	C

Explanations:	
A.	First part is plausible since CIAS does close CIVs, however the NC CIVs close on a CSAS actuation. Second part is correct.
B.	First part is plausible since CIAS does close CIVs, however the NC CIVs close on a CSAS actuation. Second part is plausible since the NC CIVs are closed during an intersystem LOCA to minimize the amount of reactor coolant which escapes containment, however a CSAS actuates on high containment pressure which would not occur on an intersystem LOCA.
C.	Correct.
D.	First part is correct. Second part is plausible since the NC CIVs are closed during an intersystem LOCA to minimize the amount of reactor coolant which escapes containment, however a CSAS actuates on high containment pressure which would not occur on an intersystem LOCA.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	8	
Reference Provided:	N	
Learning Objective:	Describe what automatically initiates the Containment Spray Actuation System (CSAS) and its function.	

Technical Reference:	40AO-9ZZ17, Inadvertent PPS-ESFAS Actuations

Attachment C-7 CSAS Train A

Actuation Leg		Component	Handswitch	Actuated Condition	In Actuated Condition (Circle one)	As Left Condition (Circle one)
1-3		Diesel Generator A	DGA-HS-1	Running	Y / N	Run / Stop
1-3		Control Room Essential AHU Fan A	HJA-HS-28	Running	Y / N	Run / Stop
1-3		Essential Chiller / Chilled Water Pump A	ECA-HS-1A	Running	Y / N	Run / Stop
1-3		Essential Cooling Water Pump A	EWA-HS-1	Running	Y / N	Run / Stop
1-3		Essential Spray Pond Pump A	SPA-HS-1	Running	Y / N	Run / Stop
	2-4	Containment Spray A Discharge to Spray Header 1 Valve	SIA-HS-672	Open	Y / N	Open / Closed
1-3		HPSI Pump A	SIA-HS-1	Running	Y / N	Run / Stop
1-3		Containment Spray Pump A	SIA-HS-5	Running	Y / N	Run / Stop
1-3		LPSI Pump A	SIA-HS-3	Running	Y / N	Run / Stop
1-3		RCP Control Bleed-Off Header to VCT Isolation Valve	CHA-HS-506	Closed	Y / N	Open / Closed
1-3		NCW Containment Downstream Return Isolation Valve	NCA-HS-402	Closed	Y / N	Open / Closed
1-3		Instrument Air Outside Containment Isolation Valve	IAA-HS-2	Closed	Y / N	Open / Closed

Technical Reference:	40AO-9ZZ17, Inadvertent PPS-ESFAS Actuations

Attachment C-8 CSAS Train B

Actuation Leg		Component	Handswitch	Actuated Condition	In Actuated Condition (Circle one)	As Left Condition (Circle one)
1-3		Diesel Generator B	DGB-HS-2	Running	Y / N	Run / Stop
1-3		Control Room Essential AHU Fan B	HJB-HS-29	Running	Y / N	Run / Stop
1-3		Essential Chiller / Chilled Water Pump B	ECB-HS-2A	Running	Y / N	Run / Stop
1-3		Essential Cooling Water Pump B	EWB-HS-2	Running	Y / N	Run / Stop
1-3		Essential Spray Pond Pump B	SPB-HS-2	Running	Y / N	Run / Stop
	2-4	Containment Spray B Discharge to Spray Header 2 Valve	SIB-HS-671	Open	Y / N	Open / Closed
1-3		HPSI Pump B	SIB-HS-2	Running	Y / N	Run / Stop
1-3		Containment Spray Pump B	SIB-HS-6	Running	Y / N	Run / Stop
1-3		LPSI Pump B	SIB-HS-4	Running	Y / N	Run / Stop
1-3		Essential Electric Auxiliary Feed Pump	AFB-HS-10	Running	Y / N	Run / Stop
1-3		RCP Control Bleed-Off Header to VCT Isolation Valve	CHB-HS-505	Closed	Y / N	Open / Closed
1-3		NCW Containment Upstream Return Isolation Valve	NCB-HS-403	Closed	Y / N	Open / Closed
1-3		NCW Containment Upstream Supply Isolation Valve	NCB-HS-401	Closed	Y / N	Open / Closed

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Instrument Air: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation	**Tier**	**1**		
	Group	**1**		
	K/A	**065 G 2.1.7**		
	IR	**4.4**		

Question 17

Per 40AO-9ZZ06, Loss of Instrument Air, as IA pressure degrades, power production stability BEGINS to be affected when pressure drops below ___(1)___ psig, and the MOST unstable IA pressure plateau affecting Reactor operation occurs at an IA header pressure of approximately ___(2)___ psig.

A. (1) 90
   (2) 50

B. (1) 90
   (2) 40

C. (1) 70
   (2) 50

D. (1) 70
   (2) 40

Proposed Answer:	D

Explanations:	
A.	First part is plausible since this is the pressure at which the first IA component fails, however this will only bypass the Demins and will not impact power production. Second part is plausible as this is the pressure at which MSIVs and FW Economizer valves fail, however those valves all fail as-is.
B.	First part is plausible since this is the pressure at which the first IA component fails, however this will only bypass the Demins and will not impact power production. Second part is correct.
C.	First part is correct. Second part is plausible as this is the pressure at which MSIVs and FW Economizer valves fail, however those valves all fail as-is.
D.	Correct.

Question Source:	x	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	x	Comprehension or Analysis

Level of Difficultly:	4	
10CFR55.41:	4	
Reference Provided:	N	
Learning Objective:	Determine the major effects on plant operation as instrument air pressure degrades.	

Technical Reference:	LOIT Loss of Instrument Air Lesson Plan

EO: 1.2 Identify who decides when a reactor trip is required.

Introduction

The Loss of Instrument Air procedure provides guidance on when to consider action, in the form of a note that informs the CRS that IA pressure of less than 70 psig will affect plant power production stability and that the most unstable IA pressure plateau affecting reactor operation will be 40 psig where Letdown, Reactor makeup water, Charging, Pressurizer Spray and Reactor Drain Tank are impacted.

Main Idea

Considering the information provided by the procedure and the actual conditions in the plant, the CRS will decide if either a trip of the Reactor or a downpower to a more stable condition is needed, or if maintaining current conditions is the right thing to do.

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Generator Voltage and Electric Grid Disturbances: Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Turbine / Generator control	**Tier**	**1**	[redacted]	
	Group	**1**	[redacted]	
	K/A	**077 AK2.07**		
	IR	**3.6**	[redacted]	

Question 18

Given the following conditions:

- Unit 1 is operating at 100% power
- The Main Generator is bucking 150 MVAR

Subsequently:

- A grid disturbance has resulted in the Main Generator now bucking only 100 MVAR
- The ECC has directed Unit 1 to take action to resume bucking 150 MVAR

In order to accomplish this, the crew will need to ___(1)___ Main Generator ___(2)___ .

A. (1) raise
   (2) voltage

B. (1) raise
   (2) frequency

C. (1) lower
   (2) voltage

D. (1) lower
   (2) frequency

Proposed Answer:	C

Explanations:	
A.	First part is plausible since a load adjustment is required in order to resume initial loading, however in this case, load must be lowered. Second part is correct.
B.	First part is plausible since a load adjustment is required in order to resume initial loading, however in this case, load must be lowered. Second part is plausible since adjusting frequency will change load, however that would change real load, not reactive load.
C.	Correct.
D.	First part is correct. Second part is plausible since adjusting frequency will change load, however that would change real load, not reactive load.

Question Source:	x	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	x	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	4	
Reference Provided:	N	
Learning Objective:	Describe how the Main Generator Excitation and Voltage Regulation System (EX2100e and MarkVIe) supports the operation of the following systems: Main Generator	

Technical Reference:	LOIT Main Generator Excitation and Regulation Lesson Plan

REACTIVE POWER

Reactive Power – The power EXCHANGED between the source and load due to the expansion and collapse of magnetic (inductive) and electrostatic (capacitive) fields.

Units of reactive power are: Vars (Volt Amperes Reactive), K Vars (KiloVars), MVars (MegaVars).

Technical Reference:	LOIT Main Generator Excitation and Regulation Lesson Plan

EO: 1.19 State the meanings of VARs out, VARS in, +Vars, -Vars, boost VARs, buck VARs, lagging VARs, and leading VARS.

Introduction
Several different terms/designators are used to identify the type of and magnitude of reactive power that a generator is carrying. The terms presented here are used by personnel in the control rooms and ECC.

Main Idea
COMMON TERMS USED WHEN REFERRING TO REACTIVE POWER (VARS).

VARS when current is actually lagging the voltage.

- VARS out.
- Positive VARS (+VARS).
- Lagging VARS.
- Boost.

VARS when current is actually leading the voltage.

- VARS in.
- Negative VARS (-VARS).
- Leading VARS.
- Buck.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Dropped Control Rod: Ability to determine and interpret the following as they apply to the Dropped Control Rod: Rod position indication to actual rod position	**Tier**	**1**		
	Group	**2**		
	K/A	**003 AA2.01**		
	IR	**3.7**		

Question 19

Given the following condition:

- Unit 1 was operating at 100% power when CEA 15, a Group 5 CEA, slipped 50” into the core

Prior to any operator actions being taken, which of the following will provide an ACCURATE CURRENT POSITION of the slipped CEA?

1. Core Monitoring Computer
2. Core Protection Calculators
3. Selected CEA Position indication on B04 (if selected to CEA 15)

A. 1 ONLY

B. 2 ONLY

C. 1 and 3 ONLY

D. 2 and 3 ONLY

Proposed Answer:	B

Explanations:	
A.	Plausible since rod position is tracked in the CMC, however following a slipped CEA, this indication is not accurate
B.	Correct.
C.	Plausible since rod position is tracked in the CMC and by Selected CEA Position indication, however following a slipped CEA, these indications are not accurate
D.	Plausible since rod position can be displayed using the Selected CEA Position indication on B04, however following a slipped CEA, this indication is not accurate

Question Source:	x	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		**Memory or Fundamental Knowledge**
	x	**Comprehension or Analysis**

Level of Difficultly:	**3**	
10CFR55.41:	**6**	
Reference Provided:	**N**	
Learning Objective:	**Describe the function of the CEA Position inputs to the Core Protection Calculators.**	

Technical Reference:	LOIT CEDMCS Lesson Plan

Since CPCs identify rod height using reed switches, it will still indicate correctly following a slipped CEA

Core Protection Calculators (CPCs)

Provides CWP signal (CEA Withdrawal Prohibit) to CEDMCS

CEDMCS provides RSPT (target CEAs) input to the CPCs

Technical Reference:	LOIT CEDMCS Lesson Plan

commands the selected CEA or Group of CEAs to move in the appropriate direction

- Individual CEA Selection switches
  - 2 rotary switches, one labeled TENS and one labeled UNITS; any one individual CEA of a selected group can be designated for motion with the joystick; also selects which CEA pulse position is used by the "Selected CEA Position" digital display

Technical Reference:	40AO-9ZZ11, CEA Malfunctions

Since the Core Monitoring Computer (CMC) uses pulse counters for CEA position indication, following a slipped CEA, the actual position of the slipped CEA must be manually inputted to restore accurate CEA position indication in the CMC.

3.0 DROPPED OR SLIPPED CEA MODE 1 OR 2

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 44. PERFORM Appendix D, Resetting COLSS and PMS CEA Positions, to reset **ANY** of the following as needed: • PC CEA position • CMC CEA position • COLSS	

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Emergency Boration: Knowledge of the interrelations between Emergency Boration and the following: When use of manual boration valve is needed	**Tier**	**1**		
	Group	**2**		
	K/A	**024 AA2.02**		
	IR	**3.9**		

Question 20

Given the following conditions:

- Unit 2 has tripped from 100% power
- A boration is required to meet Reactivity Control acceptance criteria in SPTAs

Assuming depressurization of the RCS for HPSI injection is NOT desired, which ONE of the following conditions or failures would require the use of local-manual valve operation in order to borate the RCS?

A. Refueling Water Tank level of 65%

B. A loss of BOTH Boric Acid Makeup Pumps

C. Boric Acid Flow Controller, CHN-FIC-210Y, fails to zero output

D. Makeup the CHRG PMPS (VCT Bypass), CHN-UV-527, is seized closed

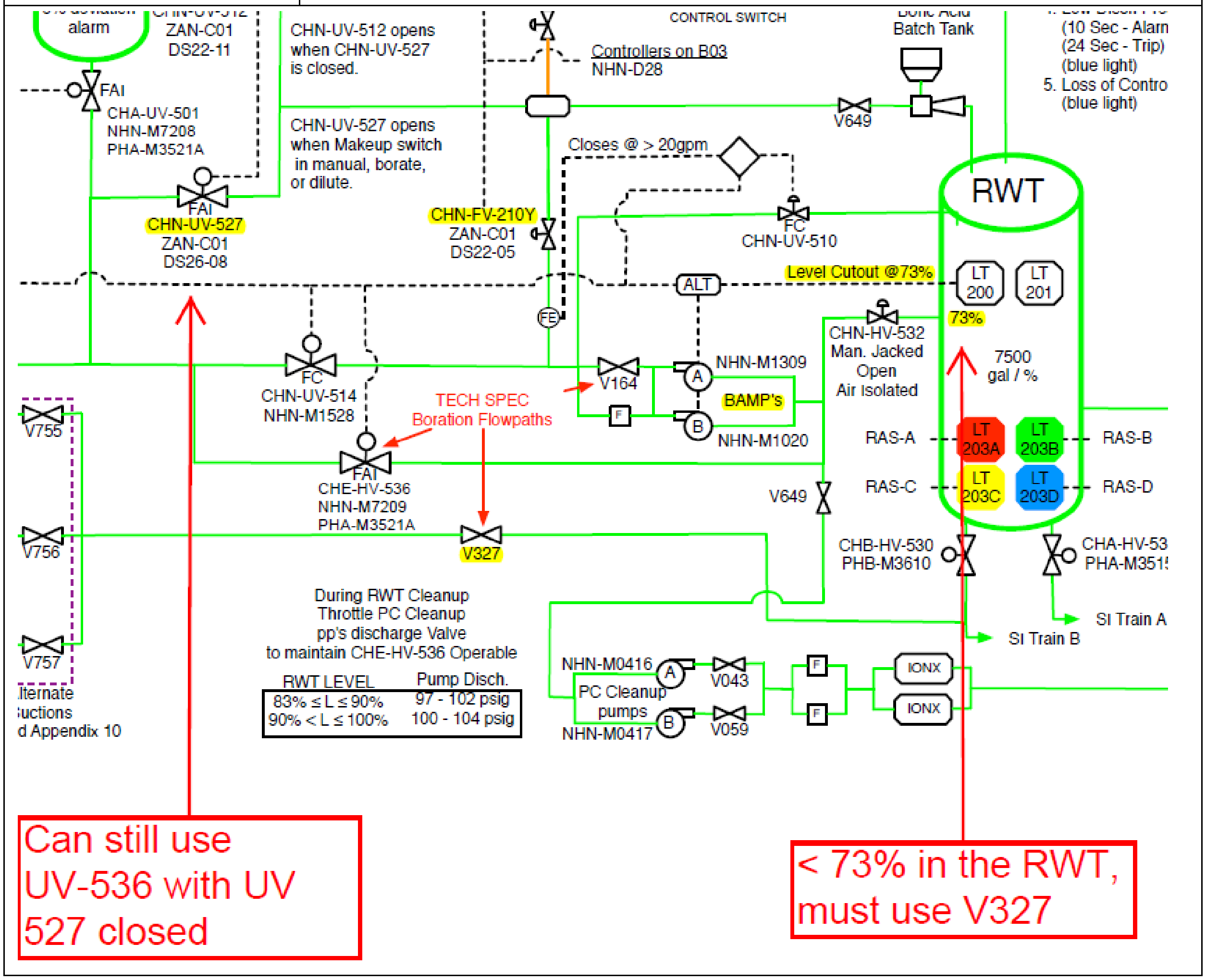
Proposed Answer:	**A**

Explanations:	
A.	Correct.
B.	Plausible since the normal boration flowpath utilizes at least one Boric Acid Makeup Pump, however in this condition a boration may still be performed using Appendix 103-D using CHE-HV-536, and all actions can be taken from the Control Room
C.	Plausible since the normal boration flowpath goes through CHN-FV-210Y (controlled by CHN-FIC-210Y), however if this controller is not available, the boration may still be achieved from the Control Room using Appendix 103-D using CHE-HV-536.
D.	Plausible since the normal boration flowpath goes through CHN-UV-527, however if this valve is not available, the boration may still be achieved from the Control Room using Appendix 103-D using CHE-HV-536.

Question Source:		**New**	
	X	**Bank**	
		Modified	
	X	**Previous NRC Exam**	**2016 NRC Q20**

Cognitive Level:		**Memory or Fundamental Knowledge**
	X	**Comprehension or Analysis**

Level of Difficultly:	**3**	
10CFR55.41:	**3**	
Reference Provided:	**N**	
Learning Objective:	**Describe the Control Room controls and indications associated with the Boric Acid Makeup Pumps**	


alarm
ZAN-C01
DS22-11
FAI
CHA-UV-501
NHN-M7208
PHA-M3521A
FAI
CHN-UV-527
ZAN-C01
DS26-08
CHN-UV-512 opens
when CHN-UV-527
is closed.
CHN-UV-527 opens
when Makeup switch
in manual, borate,
or dilute.
CONTROL SWITCH
Controllers on B03
NHN-D28
Closes @ > 20gpm
CHN-FV-210Y
ZAN-C01
DS22-05
FE
Batch Tank
V649
(10 Sec - Alarm
(24 Sec - Trip)
(blue light)
5. Loss of Contro
(blue light)
RWT
FC
CHN-UV-510
ALT
Level Cutout @73%
LT 200
LT 201
73%
7500
gal / %
CHN-HV-532
Man. Jacked
Open
Air Isolated
FC
CHN-UV-514
NHN-M1528
TECH SPEC
Boration Flowpaths
V164
NHN-M1309
BAMP's
NHN-M1020
F
A
B
RAS-A
RAS-B
RAS-C
RAS-D
LT 203A
LT 203B
LT 203C
LT 203D
V755
V756
V757
FAI
CHE-HV-536
NHN-M7209
PHA-M3521A
V327
V649
CHB-HV-530
PHB-M3610
SI Train A
SI Train B
During RWT Cleanup
Throttle PC Cleanup
pp's discharge Valve
to maintain CHE-HV-536 Operable
RWT LEVEL
Pump Disch.
83% ≤ L ≤ 90%
97 - 102 psig
90% < L ≤ 100%
100 - 104 psig
NHN-M0416
V043
PC Cleanup
pumps
NHN-M0417
V059
IONX
IONX
d Appendix 10
Can still use
UV-536 with UV
527 closed
< 73% in the RWT,
must use V327

Technical Reference:	Appendix 103, RCS Makeup / Emergency Boration

All options to borate from the control room require RWT level > 65%

APPENDIX 103: RCS MAKEUP / EMERGENCY BORATION	Page 2 of 20

INSTRUCTIONS — CONTINGENCY ACTIONS

NOTE

KEY OPERATOR ACTION - Perfect performance of this Appendix will significantly reduce plant risk.

____ 1. PERFORM ANY of the following Attachments based on current plant conditions:

Normal Boration Path

- Attachment 103-A

CHN-UV-514

- Attachment 103-B
  - RWT > 73%
  - BAMP available
- Attachment 103-C
  - RWT > 73%
  - BAMP NOT available

(continue)

Technical Reference:	Appendix 103, RCS Makeup / Emergency Boration

All options to borate from the control room require RWT level > 65%

APPENDIX 103: RCS MAKEUP / EMERGENCY BORATION

INSTRUCTIONS | CONTINGENCY ACTIONS

____ 1. (continued)

CHE-HV-536

- Attachment 103-D
  - RWT > 73%
  - PC Cleanup Pump NOT aligned to RWT
- Attachment 103-E
  - RWT > 92%
  - PC Cleanup Pump Recircing RWT
- Attachment 103-F
  - 83%< RWT <92%
  - PC Cleanup Pump Recircing RWT

End of Appendix

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Intermediate Range Nuclear Instrumentation: Ability to locate and operate components, including local controls	Tier	1		
	Group	2		
	K/A	033 G 2.1.30		
	IR	4.4		

Question 21

During a Reactor startup, the Startup Channel NIs are procedurally directed to be removed from service AS SOON AS ___(1)___ CPS is exceeded

Per 40OP-9ZZ03, Reactor Startup, Turning Off Startup Channel NIs requires component manipulation at the ___(2)___ .

A. (1) 2,000
(2) Startup and Control Channel Drawer ONLY

B. (1) 2,000
(2) Startup and Control Channel Drawer AND Board 4

C. (1) 10,000
(2) Startup and Control Channel Drawer ONLY

D. (1) 10,000
(2) Startup and Control Channel Drawer AND Board 4

Proposed Answer:	D

Explanations:	
A.	First part is plausible as this was the value at which SU channel NIs were removed from service until a few years ago, however now they are left in service until 10,000 cps. Second part is plausible since the Startup Channel NI controls are located at the NI Cabinet, however the process for taking startup channels out of service also ensures that Control Channel NIs are energized and indicated in the control room which is done at Board 4.
B.	First part is plausible as this was the value at which SU channel NIs were removed from service until a few years ago, however now they are left in service until 10,000 cps. Second part is correct.
C.	First part is correct. Second part is plausible since the Startup Channel NI controls are located at the NI Cabinet, however the process for taking startup channels out of service also ensures that Control Channel NIs are energized and indicated in the control room which is done at Board 4.
D.	Correct.

Question Source:	x	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	x	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	Explain the operation of the Boron Dilution Alarm System (BDAS) under normal operating conditions.	

Response Section

4A12A

SU AND CNTRL CH TRBL

Startup and Control Channel Trouble

Point ID	Description	Setpoint
SEJS1	Start-Up Control Channel 1 Hi Counts Per Second	1×10^4 CPS
SEJS2	Start-Up Control Channel 2 Hi Counts Per Second	1×10^4 CPS

AUTOMATIC ACTION

- None

MANUAL ACTIONS

___ 1. **IF** this alarm is received during performance of a Reactor startup, **THEN** REFER TO the appendix for checking NI overlap and Turning off Startup Channels in ONE of the following procedures:

- ____ 40OP-9ZZ02, Initial Reactor Startup Following Refuelings
- ____ 40OP-9ZZ03, Reactor Startup

Technical Reference:	40OP-9ZZ03, Reactor Startup

Reactor Startup	40OP-9ZZ03	Revision 64

___ D. Perform the following to withdraw CEAs:

___ 1. Monitor ALL of the following closely during withdrawal:

____ • Startup/Control Channel Recorders

____ • CEAPDS Video Display

____ • Log Power Recorders

____ • Log Power Meters

___ 2. Withdraw CEAs in Manual Sequential, as directed by the Reactivity Manager, to the Target CEA Position.

___ 3. **IF** the High CPS alarm is received, **THEN** perform the following:

____ a) Complete the CEA withdrawal.

____ b) Perform Appendix D - Checking NI Overlap and Turning Off Startup Channels.

____ c) Direct Reactor Engineering to re-normalize the 1/M plot.

Technical Reference:	40OP-9ZZ03, Reactor Startup

The initial steps (section 2.2) are performed at B04 in the control room, the following steps (section 2.3, 2.4) are performed at the NI cabinet

___ 2.2 Perform the following in preparation for de-energizing the Startup Channels:

___ 2.2.1 Position SEN-HS-5A, CONTROL/STARTUP CHANNEL 1 SELECTOR, to CONT CHAN 1.

___ 2.2.2 Position SEN-HS-6A, CONTROL/STARTUP CHANNEL 2 SELECTOR, to CONT CHAN 2.

___ 2.2.3 Shift SEN-JR-5, to display the Control Channels, using the arrows on the circular selector on the face.

___ 2.3 Perform the following at the Startup and Control Channel Drawer, to turn off High Voltage:

___ 2.3.1 Press the H.V. PERMIT push-button on Startup Channel 1.

___ 2.3.2 Observe the amber H.V. PERMIT light is off, at Startup Channel 1.

___ 2.3.3 Press the H.V. PERMIT push-button on Startup Channel 2.

___ 2.3.4 Observe the amber H.V. PERMIT light is off, at Startup Channel 2.

___ 2.4 Perform the following at the Startup and Control Channel Drawer, to shift indication to the Control Channels:

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Fuel Handling Incidents: Knowledge of the interrelations between the Fuel Handling Incidents and the following: Radiation monitoring equipment (portable and installed)	Tier	1		
	Group	2		
	K/A	036 AK2.02		
	IR	3.4		

Question 22

Given the following conditions:

- Unit 2 is in MODE 6
- Fuel off-load is in progress
- An irradiated fuel assembly has been dropped in the vessel resulting in rising radiation levels inside Containment

If Containment radiation levels continue to rise, a CPIAS actuation will be initiated from which of the following Containment radiation monitors?

A. RU-1, Containment Building Atmosphere Monitor

B. RU-33, Refueling Machine Area Monitor

C. RU-34, Containment Building Refueling Purge Exhaust Monitor

D. RU-37, Power Access Purge Area Monitor

Proposed Answer:	**D**

Explanations:	
A.	Plausible since RU-1 is a TS required rad monitor located inside containment, and has connections to the PPS system (RU-1 isolates on a CIAS), however RU-1 does not actuate a CPIAS during a fuel handling accident inside containment.
B.	Plausible since RU-33 is mounted next to the refueling cavity and is installed specifically to monitor radiation levels during refueling activities, however RU-33 does not acutate CPIAS in the event of a fuel handling accident inside containment.
C.	Plausible since RU-34 is a purge exhaust monitor that it would be the RM to actuate a containment purge isolation signal, however RU-34 does not actuate CPIAS during a fuel handling accident inside containment.
D.	Correct.

Question Source:	**x**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		**Memory or Fundamental Knowledge**
	x	**Comprehension or Analysis**

Level of Difficultly:	**3**	
10CFR55.41:	**11**	
Reference Provided:	**N**	
Learning Objective:	**Describe controls, actuations and interlocks associated with the Radiation Monitoring System.**	

Technical Reference:	Radiation Monitor System Tech Manual

2.3 Containment Building Atmosphere Monitor, Channel "B" (CBB) SQB-RU-1

The purpose of the containment building atmosphere radiation monitor is to continuously monitor the containment building atmosphere radioactivity levels as an indication of reactor coolant pressure boundary (RCPB) leakage in accordance with regulatory guide 1.45. Reg Guide 1.45 only addresses the use of the particulate and gas channels for leakage detection. Exception is taken to Reg Guide 1.45 in section 1.8 of the updated FSAR, as there is no way to relate monitor readings in terms of RCPB leakage flow rate. This exception states that the radiation monitor will be used as a qualitative indication only of an increase in the RCPB leakage.

Although this monitor is included as part of the SRMS, it is not required to meet IE qualification requirements as described in IEEE standard 323-1974. However, because the particulate channel of the monitor is required to survive an safe shutdown earthquake (SSE), this channel is qualified seismic category I. The monitor is equipped with a RIC module to provide monitor control and indication in the event of DCU failure. The RIC is mounted in the SRMS panel in the control room with the balance of the safety related monitor RIC units. Because it does not meet the separation requirements from the qualified RIC units, the CBB RIC must meet the requirements for "associated equipment" in accordance with IEEE standard 384-1977.

The monitor was originally equipped with a hygrometer channel. This channel was not necessary to meet the requirements of regulatory guide 1.45 and was removed. The particulate channel of the monitor was equipped with a moving filter mechanism but was replaced with a fixed filter unit.

The CBB monitor is located just outside the containment building. It samples the containment atmosphere through piping penetrations. The sampling point inside containment is located on the operating level between two of the normal cooling unit intakes. This location facilitates RCPB leak detection.

Isolation valves at the penetration automatically shut to isolate the monitor from pressure and temperature transients following a loss of coolant accident (LOCA). These valves do not shut until receipt of a containment isolation actuation signal (CIAS). This allows the monitor to function properly during an event where only a slight over pressure is applied to the sampler piping. RU-1 is rated for 10 psig.

Technical specification apply. Required monitor features for operability are the particulate and noble gas channels, the sample pumping system, and the RIC module. The particulate and noble gas channels can be considered separately from an operability standpoint. The flow control system itself is not a required feature. However, because the monitor obtains its sample flow input used in the activity calculation and low sample flow annunciation from the flow control module, the flow measurement portion of the flow control system is required (see drawing 13-M-HCP-001).

Technical Reference:	Radiation Monitor System Tech Manual

2.29 Refueling Machine Area Monitor, (RMAA) SQA-RU-33

The RMAA radiation monitor continuously monitors radiation levels in the refueling cavity area of the containment building. The detector is wall mounted overlooking the cavity. The primary function of this monitor is to provide local indication of abnormally high radiation levels in the event of a fuel handling accident or accidental criticality in containment. This monitor meets the requirements of 10CFR70.24 and regulatory guide 8.12. An RIA module is provided for local indication and alarm in the monitored near the detector. This monitor is normally used only when refueling operations are in progress. RU-33 is usually down powered and stored in a low radiation area during plant power operations.

Although this monitor is included as part of the SRMS, it is not required to meet IE qualification requirements as described in IEEE standard 323-1974 as it has no safety function. The monitor is equipped with a RIC module to provide monitor control and indication in the event of DCU failure. The RIC is mounted in the SRMS panel in the control room with the balance of the safety related monitor RIC units. Because it does not meet the separation requirements from the qualified RIC units, the RMAA RIC must meet the requirements for "associated equipment" in accordance with IEEE standard 384-1977 (see drawing 13-M-CPP-001).

The actual engineered safety feature actuation function of isolating the containment purge exhaust (CPIAS initiation) is provided by the power access purge area monitors (PAPA and PAPB) SQA-RU-37 and SQB-RU-38.

This monitor has no operability requirements per the technical specifications.

Technical Reference:	Radiation Monitor System Tech Manual

2.30 Containment Building Refueling Purge Exhaust Monitor, (CBPB) SQB-RU-34

The CBPB radiation monitor continuously monitors noble gas radioactivity levels in the containment building purge exhaust. The primary function of this monitor is to provide a high activity alarm associated with abnormal radioactivity levels in the purge exhaust. Although this monitor is included as part of the SRMS, it is not required to meet IE qualification requirements as described in IEEE standard 323-1974 as it has no safety function. The monitor is equipped with a RIC module to provide monitor control and indication in the event of DCU failure. The RIC is mounted in the SRMS panel in the control room with the balance of the safety related monitor RIC units. Because it does not meet the separation requirements from the qualified RIC units, the CBB RIC must meet the requirements for "associated equipment" in accordance with IEEE standard 384-1977 (see drawing 13-M-CPP-001).

The actual engineered safety feature actuation function of isolating the containment purge exhaust (CPIAS initiation) is provided by the power access purge area monitors (PAPA and PAPB) SQA-RU-37 and SQB-RU-38.

This monitor has no operability requirements per the technical specifications and the actual effluent radiation monitor is the plant vent monitor.

2.31 Power Access Purge Area Monitors, SQA-RU-37 (PAPA) and SQB-RU-38 (PAPB)

The PAPA and PAPB are located outside the containment between the power access purge exhaust and refueling purge exhaust ducts. These channels monitor the ducts for purged airborne radioactivity concentrations that could potentially result in an offsite dose exceeding 10CFR100 limits. The primary function of these monitors is to provide a engineered safety feature actuation on a high-high alarm initiating containment building purge supply and exhaust isolation (CPIAS). These monitors are included as part of the SRMS, and meet IE qualification requirements as described in IEEE standard 323-1974. These monitors are equipped with a RIC module to provide monitor control and indication in the event of DCU failure. The RIC is mounted in the SRMS panel in the control room with the balance of the safety related monitor RIC units (see drawing 13-M-CPP-001). Technical specifications apply. Required monitor features for operability are the area monitor itself, its associated ESF actuation capability, and control room indication and annunciation.

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Accidental Liquid Radwaste Release: Knowledge of the reasons for the following responses as they apply to the Accidental Liquid Radwaste Release: Actions contained in EOP for accidental liquid radioactive-waste release	**Tier**	**1**	[redacted]	
	Group	**2**	[redacted]	
	K/A	**059 AK3.04**		
	IR	**3.8**	[redacted]	

Question 23

During a SGTR concurrent with an unisolable ESD on the same SG, the affected SG should be fed at a MAXIMUM feedrate of ___(1)___ until the U-tubes are covered in order to minimize ___(2)___ .

A. (1) 1000 gpm
(2) the thermal stresses on the SG U-tubes

B. (1) 1000 gpm
(2) the magnitude of the radioactive release

C. (1) 1600 gpm
(2) the thermal stresses on the SG U-tubes

D. (1) 1600 gpm
(2) the magnitude of the radioactive release

Proposed Answer:	D

Explanations:	
A.	First part is plausible as this is the maximum allowable feedrate when feeding a dry SG, however when feeding a faulted/ruptured SG, the maximum allowable feedrate is 1600 gpm. Second part is plausible since the uncovery and recovering of the SG u-tubes would put additional thermal stresses on the tubes and doing so could further degrade the tubes, however the primary reason for covering the tubes is to dilute the reactor coolant which is entering the SG to reduce the magnitude of the radioactive release.
B.	First part is plausible as this is the maximum allowable feedrate when feeding a dry SG, however when feeding a faulted/ruptured SG, the maximum allowable feedrate is 1600 gpm. Second part is correct.
C.	First part is correct. Second part is plausible since the uncovery and recovering of the SG u-tubes would put additional thermal stresses on the tubes and doing so could further degrade the tubes, however the primary reason for covering the tubes is to dilute the reactor coolant which is entering the SG to reduce the magnitude of the radioactive release.
D.	Correct.

Question Source:	x	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	x	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	12	
Reference Provided:	N	
Learning Objective:	Given that a SGTR is in progress with a coincident ESD and specific SG and RCS temperatures, levels and pressures, determine which SG will be used to maintain RCS heat removal in accordance with 40EP-9EO09.	

INSTRUCTIONS	CONTINGENCY ACTIONS

* 15. IF the Steam Generator with the tube rupture has ANY of the following indications of an ESD:

- Abnormal steam generator pressures
- Abnormal steam generator levels
- Abnormal RCS cold leg temperatures

AND it is uncontrollably steaming to atmosphere,
THEN ensure at least ONE of the following conditions is met:

- The affected Steam Generator has level being restored by feedwater flow 1360 - 1600 gpm (0.8 - $0.92X10^6$ lbm/hr)
- The affected Steam Generator has level 45 - 60% [45 - 60%] NR with feedwater available to maintain level

Technical Reference:	Appendix 44, Feeding with the Condensate Pumps

____ 14. **IF** Steam Generator #1 was selected, **THEN** perform the following:

a. Fast close Steam Generator #1 MSIVs by using the following pushbuttons:

- SGA-HS-251
- SGB-HS-253

b. Lower Steam Generator #1 pressure below the condensate pump discharge pressure using SG 1 ADVs.

b.1 PERFORM Appendix 18, Local ADV Operation.

c. Maintain Steam Generator #2 pressure less than 1200 psia.

d. **IF** Steam Generator #1 is dry, **THEN** maintain feed flow rate of less than or equal to 1000 gpm (0.5×10^6 lbm/hr).

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: High Reactor Coolant Activity: Ability to operate and / or monitor the following as they apply to the High Reactor Coolant Activity: Failed fuel-monitoring equipment	Tier	1		
	Group	2		
	K/A	076 AA1.04		
	IR	3.2		

Question 24

While operating at power, high reactor coolant activity indicative of potential fuel cladding failure is primarily monitored by ___(1)___ , and a high alarm on this(these) radiation monitor(s) ___(2)___ automatically isolate letdown.

A. (1) Primary Coolant Activity Monitors, RU-150/151
(2) WILL

B. (1) Primary Coolant Activity Monitors, RU-150/151
(2) will NOT

C. (1) Reactor Coolant Letdown Line Radiation Monitor, RU-155D
(2) WILL

D. (1) Reactor Coolant Letdown Line Radiation Monitor, RU-155D
(2) will NOT

Proposed Answer:	D

Explanations:	
A.	First part is plausible since RU-150/151 do monitor the activity levels of the RCS, however their primary function is to do so during post-accident conditions. Second part is plausible since high activity in the RCS would cause high radiation levels in the aux building, which would be mitigated by letdown isolating, however letdown does not auto isolate due to high RCS activity.
B.	First part is plausible since RU-150/151 do monitor the activity levels of the RCS, however their primary function is to do so during post-accident conditions. Second part is correct.
C.	First part is correct. Second part is plausible since high activity in the RCS would cause high radiation levels in the aux building, which would be mitigated by letdown isolating, however letdown does not auto isolate due to high RCS activity.
D.	Correct.

Question Source:		New	
	X	Bank	
		Modified	
	x	Previous NRC Exam	2019 NRC Q51

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	11	
Reference Provided:	N	
Learning Objective:	Explain the operation of the Letdown Process Radiation Monitor (SQN-RE-155D) under normal operating conditions.	

Technical Reference:	Radiation Monitor System Tech Manual

2.42 Primary Coolant Activity Monitors, (PCMA) SQA-RU-150 and (PCMB) SQB-RU-151

The primary coolant activity monitors consist of two independent ionization chamber channels. These monitors meet the monitoring requirements of NUREG-0737 and regulatory guide 1.97, Rev 2 for the circulating coolant activity monitors. The purpose of these monitors is to assess activity levels in the primary coolant under post accident conditions. The detectors are physically located next to a cold leg of each of the steam generators. These monitors are included as part of the SRMS, and meet IE qualification requirements as described in IEEE standard 323- 1974. These monitors are equipped with a RIC module to provide monitor control and indication in the event of DCU failure. The RIC is mounted in the SRMS panel in the control room with the balance of the safety related monitor RIC units. The micro-computers for these monitors are located in the control building to take advantage of the low radiation fields afforded by this category I structure (see drawings 13-J-ZCF-006, 13-J-ZCF-005 and 13-J-ZCF-009).

Although these monitors are part of the SRMS, they have no operability requirements per the technical specifications.

2.43 Personnel Access Monitors, SQN-RU-152, RU-153, and RU-154

These monitors provide radiation exposure rate information for various plant areas under post accident conditions. These monitors meet the monitoring requirements of NUREG-0737 and regulatory guide 1.97, Rev 2 for monitoring areas that may require access post accident. The detectors are physically located as described on the referenced drawings. These monitors are included as part of the PAMS with their micro-computers located in the PAMU. Each channel is equipped with an RIA to provide local indication and alarm in the monitored area near the detector for each channel.

These monitors have no operability requirements per the technical specifications.

2.44 Reactor Coolant Letdown Line Radiation Monitor SQN-RE-155D

This monitor is also referred to as an area monitor (ARM). The fourth available channel of penetration leakage monitor (SQN-RU-155) is used to trend the letdown primary coolant activity. This area radiation monitor provides a continuous recording in the control room of reactor coolant gross gamma activity thus providing a measure of fuel cladding integrity. A high alarm is provided in the control room. Local and remote samples in the CVCS provide the primary means for determining RCS activity. The reactor coolant letdown line monitor serves only as a trending device to warn the operator of possible fuel cladding failure. Verification of the ARM is done by grab sample measurement.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Natural Circulation: Knowledge of the operational implications of the following concepts as they apply to the (Natural Circulation Operations): Normal, abnormal and emergency operating procedures associated with (Natural Circulation Operations)	**Tier**	**1**		
	Group	**2**		
	K/A	**CE A13 AK1.2**		
	IR	**3.2**		

Question 25

Given the following conditions:

- Unit 3 is operating at 100% power
- Fast bus transfer was blocked on NAN-S01 and NAN-S02 due to low grid voltage

Subsequently:

- The reactor was tripped due to a SGTR on SG #1
- SPTAs have been completed and the CRS has entered 40EP-9EO04, SGTR
- The crew is preparing to conduct a cooldown and isolate SG #1

Procedurally, the cooldown rate limit (averaged over one hour) PRIOR to isolating SG #1 is ___(1)___ , and the cooldown rate limit AFTER SG #1 is isolated is ___(2)___ .

A. (1) 30°F/hr
(2) 30°F/hr

B. (1) 30°F/hr
(2) 100°F/hr

C. (1) 100°F/hr
(2) 30°F/hr

D. (1) 100°F/hr
(2) 100°F/hr

Proposed Answer:	C

Explanations:	
A.	Plausible that 30°F/hr would be the cooldown rate for the entire cooldown since a rapid cooldown could potentially uncouple the primary and secondary during natural circulation, however the 30°F/hr limit is only when one SG is isolated.
B.	Plausible that the cooldown rate would be limited prior to isolating the ruptured SG since we use both SGs for the initial cooldown and a 100°F/hr cooldown rate using the ruptured SG could make the tube break worsen, however the strategy is to cooldown as quick as possible to < 540°F to isolate the ruptured SG and then continue at 30°F/hr to ensure the primary and secondary do not become uncoupled with asymmetrical steaming following the SG isolation.
C.	Correct.
D.	Plausible that the cooldown rate would be unaffected following the SG isolation as this is true with forced circulation.

Question Source:		New	
	X	Bank	
		Modified	
	x	Previous NRC Exam	2016 NRC Q26

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	Given that the SGTR EOP is being implemented, describe the SGTR EOP mitigation strategy in accordance with 40EP-9EO04.	

Technical Reference:	PVNGS EOP Operations Expectations

8.0 STEAM GENERATOR TUBE RUPTURE (SGTR)

SGTR Step 2

1. It is expected that the CRS will periodically review the event with the SM to update information pertaining to the Emergency Plan which might result in changing the event classification.

SGTR Step 5

1. If SI throttle criteria is met, then SI flow should not be maximized. The contingency action to restore electrical power to valves and pumps is intended to encompass resetting thermals, relays, or manually closing breakers. It is not intended to include complicated evolutions such as the cross connection buses.

SGTR Step 10

1. The CRS should log REP CET with no RCPs running or Tc with RCPs running in the placekeeper. The secondary operator should commence a rapid cooldown to reduce Th to less than 540°F to get the ruptured SG isolated as rapidly as possible. The cooldown should not stop when 540°F is reached but should continue to SDC entry conditions at a rate not to exceed 100°F in a 1 hour period. Analysis for this event has shown that reactor vessel stresses are acceptable for a 70°F cooldown in 5 minutes followed by a 55 minute soak. So the initial cooldown may be done as rapidly as can be controlled. Logging REP CET or Tc in the placekeeper will give the operator the initial temperature reading to base the cooldown on once the cooldown rate log is begun. During all phases of the cooldown, RCS temperature and pressure should be monitored to avoid exceeding a maximum cooldown rate greater than Technical Specification Limitations. The motor driven auxiliary or main feedwater pumps should be used to reduce the release of potentially radioactive steam from turbine driven pump exhausts. If the motor driven pumps are not available, steam from the intact SG should be used to drive the turbine driven aux feed pump.

Technical Reference:	40EP-9EO04, Steam Generator Tube Rupture

At this point in the procedure, the most affected SG is already isolated, and due to the failure of FBT, the unit is in natural circulation, therefore the subsequent cooldown rate should be limited to 30°F/hr

PALO VERDE NUCLEAR GENERATING STATION **STEAM GENERATOR TUBE RUPTURE**	40EP-9EO04 Revision 33 Page 17 of 48

INSTRUCTIONS	CONTINGENCY ACTIONS
29. Direct the STA to determine that the RCS boron concentration is sufficient to maintain the reactor 1% or more shutdown during the entire cooldown.	
30. IF the RCS boron concentration is sufficient to maintain the reactor 1% or more shutdown during the entire cooldown, THEN perform the following: a. PERFORM Appendix 5, RCS and PZR Cooldown Log. b. IF a natural circulation cooldown will be performed, THEN limit the cooldown rate to approximately 30°F / hr.	

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Steam Generator Tube Leak: Knowledge of the operational implications of the following concepts as they apply to Steam Generator Tube Leak: Leak rate vs. pressure drop	**Tier**	**1**		
	Group	**2**		
	K/A	**037 AK1.02**		
	IR	**3.9**		

Question 26

Per 40DP-9AP09, SG Tube Rupture Technical Guidelines, after the most affected SG has been isolated during a SGTR…

(1) Why is the isolated SG pressure maintained less than 1135 psia?

(2) Why is D/P between the RCS and the isolated SG maintained at +/- 50 psid?

A. (1) To minimize the likelihood of lifting a MSSV on the isolated SG
(2) To minimize the leak rate to and from the affected SG

B. (1) To minimize the likelihood of lifting a MSSV on the isolated SG
(2) To minimize the pressure stress across the degraded SG U-tube(s) to prevent further degradation

C. (1) To ensure SBCS remains available by minimizing D/P across the MSIV Bypass Valve
(2) To minimize the leak rate to and from the affected SG

D. (1) To ensure SBCS remains available by minimizing D/P across the MSIV Bypass Valve
(2) To minimize the pressure stress across the degraded SG U-tube(s) to prevent further degradation

Proposed Answer:	A

Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible since lower D/P across the degraded U-tubes will lower the likelihood for further degradation, however the reason for maintaining +/- 50 psid is to minimize leak rate.
C.	First part is plausible since reopening the MSIV bypass to reduce pressure in the isolated SG is preferred over using ADVs, however the reason for maintaining pressure < 1135 psia is to prevent lifting a MSSV on the affected SG. Second part is correct.
D.	First part is plausible since reopening the MSIV bypass to reduce pressure in the isolated SG is preferred over using ADVs, however the reason for maintaining pressure < 1135 psia is to prevent lifting a MSSV on the affected SG. Second part is plausible since lower D/P across the degraded U-tubes will lower the likelihood for further degradation, however the reason for maintaining +/- 50 psid is to minimize leak rate.

Question Source:		New	
	x	Bank	
		Modified	
	x	Previous NRC Exam	2018 NRC Q23

Cognitive Level:	x	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	Given the SGTR EOP is being used and given plant conditions, determine an appropriate pressure target for depressurization and state the basis for this value in accordance with 40EP-9EO04.	

Technical Reference:	LOIT SGTR Lesson Plan

Licensed Operator Initial Training | **Page: 30 of 53**

Title: **Steam Generator Tube Rupture** | **Lesson Plan #:** **NKASMC040408**

The critical steps for this HRA are the reduction of RCS temperature to <540F and RCS pressure to <1135psia. These conditions bring reactor coolant pressure less than the lift setpoints of the MSSVs and RCS temperature to a saturation pressure less than the MSSV setpoint. This minimizes the loss of coolant to the secondary side and reduces the possibility of lifting the MSSVs and maximizes the ability to isolate the ruptured SG to stop loss of primary inventory and the release of radioactivity. Subcooling is not expected to be lost with a single tube rupture, so two RCPs will be operating. RCP operation enhances the ability to cool down the plant. Natural circulation causes cooldown to take a much longer time, but this is not critical to ultimate success if the leak rate is minimized.

Licensed Operator Initial Training | **Page: 42 of 53**

Title: **Steam Generator Tube Rupture** | **Lesson Plan #:** **NKASMC040408**

d. The SG can be allowed to cool to ambient temperatures. Best estimate analysis shows that ambient cooling could take up to 27 hours to cool the isolated Steam Generator to SDC entry conditions. Therefore, it may not be the most practical choice. During the cooldown, the operator should maintain level in the indicating range and above the tubes. Maintaining the tubes covered is desirable to minimize the transfer of radionuclides from the primary to the secondary side of the Steam Generator through pre-existing cracks. RCS pressure should be maintained approximately equal (+/- 50 psi) to the isolated Steam Generator pressure to control primary to secondary leakage and SG level.

Explanation
As each method is discussed ask the candidates for the advantages and disadvantages of each. Consider in the discussion–

- Spread of Contamination
- Effectiveness

4.5.17 Step 17 - Maintain Isolated Steam Generator Pressure

A. The intent of this step is to ensure that high pressure in the isolated Steam Generator does not cause a Main Steam Safety Valve to lift causing an uncontrolled release.

- The cooldown that was begun to isolate the affected SG should continue until SDC entry conditions are met. The plant cooldown should be sufficient to prevent the isolated SG from approaching the lift setpoint of the MSSVs.

- The MSIV Bypass Valve and the SBCS can be used to steam to the condenser. The effectiveness of the bypass valve is limited by its size. It is, however more desirable to steam the isolated SG to the condenser than to steam directly to the environment. By steaming to the condenser some of the activity will be retained in the condenser rather than being released to the environment and this will provide a monitored release path via the off gas monitor.

- However, should the pressure in an isolated steam generator approach the lift setpoint for the associated MSSVs, it is desirable from the perspective of positive operator control and minimizing the possibility of an MSSV sticking open, that the ADV open first. This is accomplished by manually opening the ADV on increasing pressure to prevent the isolated steam generator from exceeding 1135 psia, or locally opening the ADV. To minimize the unmonitored release of radioactivity, use of the atmospheric steam dump valves on the affected steam generator should be minimized.

4.5.48 Step 48 - If SIAS, Restore Systems

A. Re-energizing SIAS load shed panels is required to ensure the operability of the non-safety auxiliary feed pump, essential lighting, Containment cooling and other needed loads. Essential lighting and other needed loads can be safely restored in a controlled manner after a SIAS.

If containment level is not indicated, then normal containment cooling should be established in order to maximize recirculation of the containment atmosphere. This recirculation will minimize the possibility of local accumulations of hydrogen developing. This will also help in removing heat from the containment in order to stop containment spray as soon as possible. If containment level is indicated, normal containment cooling shall not be restored. This is due to the potential for submergence and eventual failure of the inside containment isolation valve motor operators for NC and WC.

4.5.49 Step 49 - Cooldown and Depressurize the Isolated Steam Generator

A. This step directs actions necessary to allow the RCS to be depressurized to SDC entry conditions. The pressure in an isolated Steam Generator will remain high during the cooldown due to thermal stratification of the secondary water. Without boiling and recirculation flows in the Steam Generator, the secondary side fluid is not well mixed. This pressure is a concern as the SGTR strategy maintains RCS pressure approximately equal (+/- 50 psi) to the isolated Steam Generators pressure. This strategy is intended to minimize the flow of reactor coolant through the ruptured tube(s) to the secondary side of the SG. Thus, RCS inventory is conserved, contamination of the secondary is minimized and overfilling of the isolated Steam Generator is avoided.

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Functional Recovery: Knowledge of EOP mitigation strategies	**Tier**	**1**	[redacted]	
	Group	**2**	[redacted]	
	K/A	**CE E09 G 2.4.6**		
	IR	**3.7**	[redacted]	

Question 27

Which of the following situations would REQUIRE the use of 40EP-9EO09, Functional Recovery?

1. Depressurizing a SG to restore feed with Condensate Pumps during a LOAF
2. Depressurizing the RCS to inject with LPSI Pumps during a small break LOCA with no HPSI Pumps available
3. Aligning NBN-X03 to PBB-S04 during an ESD inside Containment following a failure of NBN-X04 and ‘B’ EDG with the ‘A’ CS Pump OOS

A. 1 ONLY

B. 2 ONLY

C. 1 and 3 ONLY

D. 2 and 3 ONLY

Proposed Answer:	D

Explanations:	
A.	Plausible since use of Condensate Pumps for restoration of feedwater required use of the Functional Recovery procedure until 3-4 years ago, however now this option is contained in 40EP-9EO06, LOAF
B.	Plausible since 2 is correct, however 3 is also correct. Plausible that 3 would not require use of the FR since there is at least one train of power and CS available, however if "bus-plus" criteria is not met, the FR must be used to have a required piece of safety equipment which can be powered from its associated bus.
C.	Plausible since 3 is correct, and 1 did require use of the FR until 3-4 years ago, however that is no longer the case.
D.	Correct.

Question Source:	X	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		**Memory or Fundamental Knowledge**
	X	**Comprehension or Analysis**

Level of Difficultly:	**3**	
10CFR55.41:	**10**	
Reference Provided:	**N**	
Learning Objective:	**Given plant conditions, determine if entry into the FRP is appropriate in accordance with 40EP-9EO09.**	

Technical Reference:	40EP-9EO06, Loss of All Feedwater

LOSS OF ALL FEEDWATER

INSTRUCTIONS

6. Restore feed to at least one Steam Generator using **ANY** of the following:

AUXILARY FEEDWATER

- Appendix 38, Resetting AFA-P01
- Appendix 39, Local Operation of AFB-P01
- Appendix 40, Local Operation of AFA-P01 Using Main Steam
- Appendix 41, Local Operation of AFN-P01
- Appendix 42, Aligning Aux Feedwater Pumps Suction to RMWT
- Appendix 112, Manual Operation of AFA-P01 During a Security Event

MAIN FEEDWATER

- Appendix 43, Restarting MFPs

CONTINGENCY ACTIONS

6.1 Perform the following to establish a low pressure feedwater source:

a. **IF ALL** of the following:

- Auxiliary or Main Feedwater can **NOT** be restored
- Offsite power is available
- Feeding a Steam Generator with a condensate pump is desired

THEN PERFORM Appendix 44, Feeding with the Condensate Pumps.

b. **IF** feeding a Steam Generator with a fire pump is desired, **THEN** PERFORM Appendix 118, Cross-connect FP to AF.

6.2 **IF** feed to at least one Steam Generator can **NOT** be restored, **THEN** GO TO 40EP-9EO09, Functional Recovery to perform **ANY** of the following:

- Cross tie electrical buses to restore an Auxiliary Feedwater Pump
- Align a Condensate Pump to feed the Steam Generator(s)

Technical Reference:	40EP-9EO09, Functional Recovery

INSTRUCTIONS	CONTINGENCY ACTIONS
3. Optimize SI flow by performing the following:	
a. Check that the SI Pumps have started.	a.1 Start idle SI Pumps as necessary.
b. Check that makeup/safety injection flow is adequate. REFER TO Appendix 2, Figures.	b.1 Perform the following: 1) Ensure electrical power to valves and pumps. 2) Ensure correct control board valve lineup. 3) Ensure operation of ESF auxiliary systems. 4) Start idle Charging Pumps as needed. 5) Depressurize the RCS by controlling **ANY** of the following: • RCS Heat Removal (REFER TO the HR success path currently in use) • Pressurizer heaters and main or auxiliary spray • Charging, letdown, and HPSI flow • RCGVS using Success Path PC-2, RCGVS
While in the LOCA EOP, with no HPSI pumps and the break size not sufficient to lower pressure below LPSI Pump shutoff head, transition to the Functional Recovery is required to intentionally depressurize the RCS to initiate LPSI injection	

Technical Reference:	40EP-9EO09, Functional Recovery

9.0 MAINTENANCE OF VITAL AUXILIARIES AC

SUCCESS PATH: MVAC-1; Offsite Power

INSTRUCTIONS	CONTINGENCY ACTIONS

1. Open the Placekeeper.

NOTE

Appendix 51, Electric Plant Single Line Diagram, is available as a reference when restoring the electric plant.

---

2. **IF** at least one vital 4.16 kV AC bus is energized from a SBOG, **THEN** GO TO step 8.

3. **IF** one vital 4.16 kV AC bus is energized from offsite power, **AND** the equipment needed to maintain Safety Functions is **NOT** available on the energized bus, **THEN** perform the following to cross-tie offsite power to the de-energized bus:

   a. **IF** PBB-S04 is to be energized, **THEN** GO TO step 10.

   b. **IF** PBA-S03 is to be energized, **THEN** GO TO step 11.

This concept is referred to as "bus plus" when one bus is energized but the required equipment to mitigate the event is on the opposite train bus. This cross-tie of power is only performed in the Functional Recovery

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Reactor Coolant Pump: Ability to monitor automatic operation of the RCPS, including: Seal Injection flow	**Tier**	**2**		
	Group	**1**		
	K/A	**003 A3.01**		
	IR	**3.3**		

Question 28

Given the following conditions:

- Unit 3 is operating at 100% power
- All Seal Injection Flow Controllers are in AUTO

Subsequently:

- A failure causes CHN-PDV-240, Charging Line to RC Loop 2A Valve, to fail FULL OPEN

In response to this failure, the OUTPUT on the Seal Injection Flow Controllers will ___(1)___ in an effort to ___(2)___ Seal Injection flow.

A. (1) rise
(2) raise

B. (1) rise
(2) lower

C. (1) lower
(2) raise

D. (1) lower
(2) lower

Proposed Answer:	C

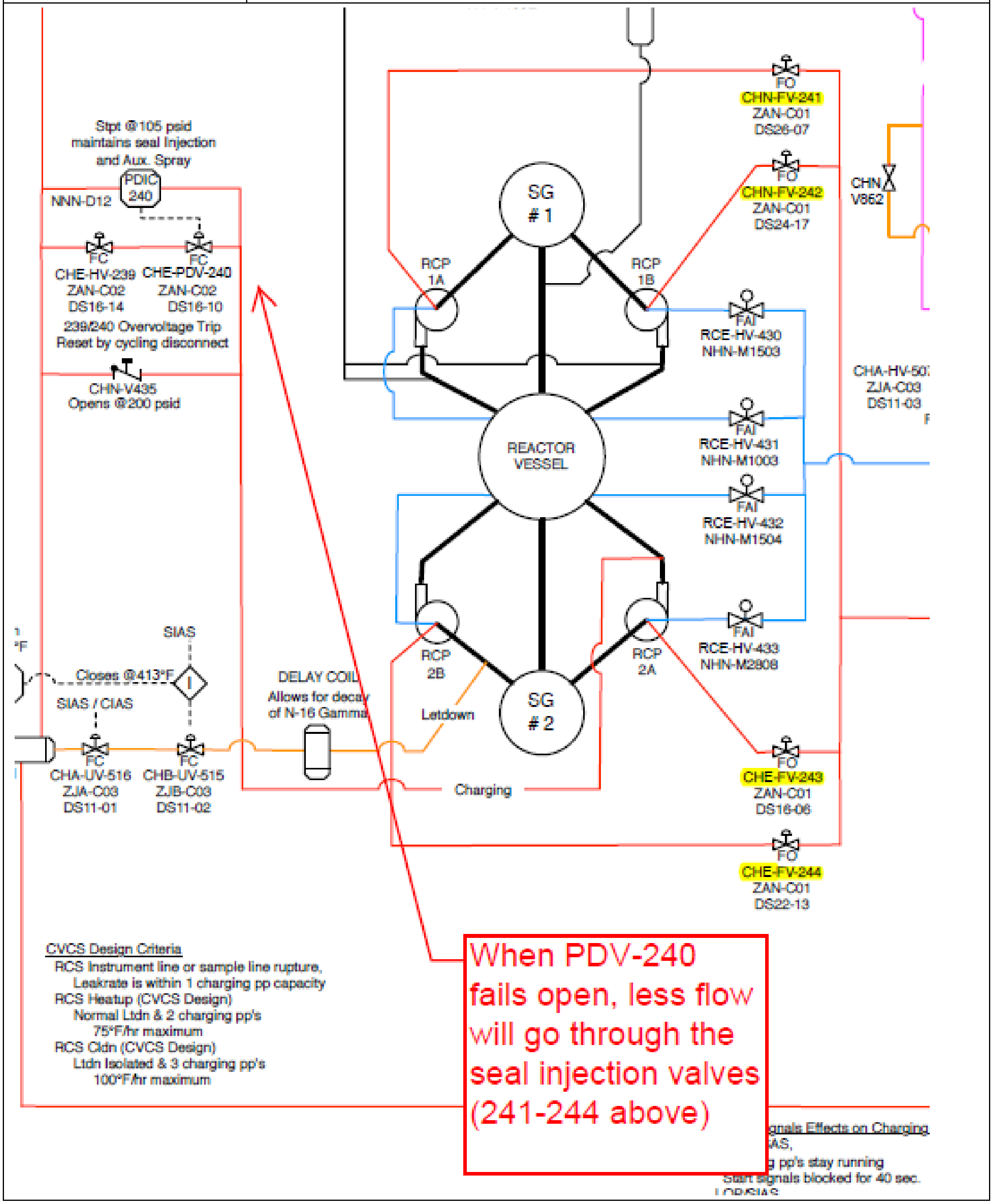
Explanations:	
A.	Plausible since flow needs to be raised following the failure, however in order to achieve this, seal injection controller output needs to lower.
B.	Plausible since the seal injection flow controllers are reverse acting, however the initial failure will cause seal injection flow to lower, not rise.
C.	Correct.
D.	Plausible since the flow controller output will lower, however this is in an effort to raise flow.

Question Source:	x	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	4	
10CFR55.41:	3	
Reference Provided:	N	
Learning Objective:	Describe the automatic functions associated with the RCP Seal Injection Isolation Valve	

Technical Reference:	PVNGS Operator Information Manual



Technical Reference:	40OP-9CH03, Reactor Coolant Pump Seal Injection System

NOTE

___ With RCS pressure less than normal operating pressure, adjustment of CHN-PDIC-240, Charging Header Backpressure Controller, may be necessary to establish the desired flowrate.

___ 6.1.8 Slowly open the seal injection to reactor coolant pump flow control valves by lowering the manual output on ALL of the following individual flow control valves. (desired flow rate is 6.0 to 7.5 gpm per pump)

- ___ CHN-FIC-241, Seal Injection to RCP 1A
- ___ CHN-FIC-242, Seal Injection to RCP 1B
- ___ CHN-FIC-243, Seal Injection to RCP 2A
- ___ CHN-FIC-244, Seal Injection to RCP 2B

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Chemical and Volume Control: Ability to monitor automatic operation of the CVCS, including: Letdown and charging flows	**Tier**	**2**	[redacted]	
	Group	**1**	[redacted]	
	K/A	**004 A3.14**		
	IR	**3.4**	[redacted]	

Question 29

Given the following conditions:

- Unit 2 is operating at 15% Reactor power
- RCN-LIC-110, Level Setpoint Control, is in LOCAL-MANUAL, with a setpoint of 40%
- RCN-HS-110, Level Control Channel X/Y Selector, is selected to ‘X’

Subsequently:

- RCA-LI-110X, Level Control Channel X, failed to 0%

With NO operator action, the Backup Charging Pump should ___(1)___ and letdown flow should ___(2)___ .

A. (1) start
(2) rise

B. (1) start
(2) remain constant

C. (1) remain off
(2) rise

D. (1) remain off
(2) remain constant

Proposed Answer:	B

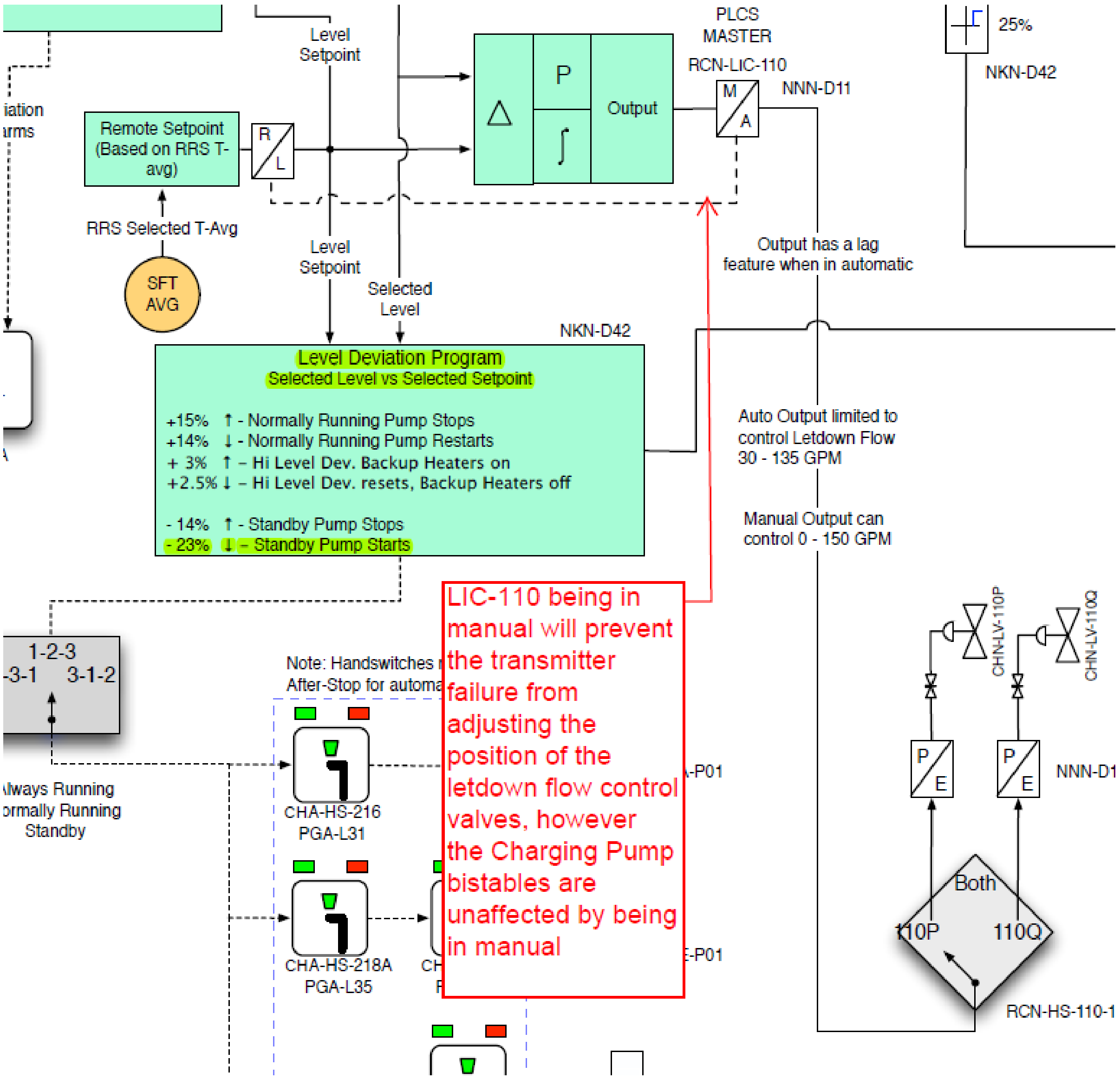
Explanations:	
A.	First part is correct. Second part is plausible since selected pressurizer level is now 40% below setpoint, however with LIC-110 in manual, letdown flow will not change.
B.	Correct.
C.	First part is plausible since the LIC-110 is in manual, however the charging pump auto start/stop bistables are not depended on the auto/manual selection on LIC-110. Second part is plausible since selected pressurizer level is now 40% below setpoint, however with LIC-110 in manual, letdown flow will not change.
D.	First part is plausible since the LIC-110 is in manual, however the charging pump auto start/stop bistables are not depended on the auto/manual selection on LIC-110. Second part is correct.

Question Source:	x	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		**Memory or Fundamental Knowledge**
	x	**Comprehension or Analysis**

Level of Difficultly:	**3**	
10CFR55.41:	**3**	
Reference Provided:	**N**	
Learning Objective:	**Describe the automatic features associated with the Pressurizer Level Control System** **Bistables.**	

Although the failure of the level transmitter would result in letdown flow adjusting to restore level to setpoint, while in manual, the valve position is only controlled by the operator. The Charging Pump bistable will function automatically even while level control is in manual.



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Residual Heat Removal: Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: RHR heat exchanger	Tier	2		
	Group	1		
	K/A	005 K6.03		
	IR	2.5		

Question 30

Given the following conditions:

- Unit 2 is in MODE 4
- SDC is in service using Train ‘A’ LPSI Pump and Train ‘A’ Auxiliaries

Subsequently:

- A tube leak occurred in the ‘A’ SDCHX

The tube leak will cause level to change in the Train ‘A’ ___(1)___ , and the crew should place the Train ‘B’ SDCHX in service per ___(2)___ .

A. (1) Spray Pond
(2) 40EP-9EO09, Functional Recovery

B. (1) Spray Pond
(2) 40EP-9EO11, Lower Mode Functional Recovery

C. (1) EW Surge Tank
(2) 40EP-9EO09, Functional Recovery

D. (1) EW Surge Tank
(2) 40EP-9EO11, Lower Mode Functional Recovery

Proposed Answer:	D

Explanations:	
A.	First part is plausible since the Spray Pond is the ultimate cooling source for the SDCHX, however the Spray Pond is used to cool the EW system which is then used to cool the SDCHX. Second part is plausible since the functional recovery procedure is used to mitigate events which initiate in MODE 3 or 4, however that is only true if LTOP is not in service and with SDC in service, LTOP is also in service.
B.	First part is plausible since the Spray Pond is the ultimate cooling source for the SDCHX, however the Spray Pond is used to cool the EW system which is then used to cool the SDCHX. Second part is correct.
C.	First part is correct. Second part is plausible since the functional recovery procedure is used to mitigate events which initiate in MODE 3 or 4, however that is only true if LTOP is not in service and with SDC in service, LTOP is also in service.
D.	Correct.

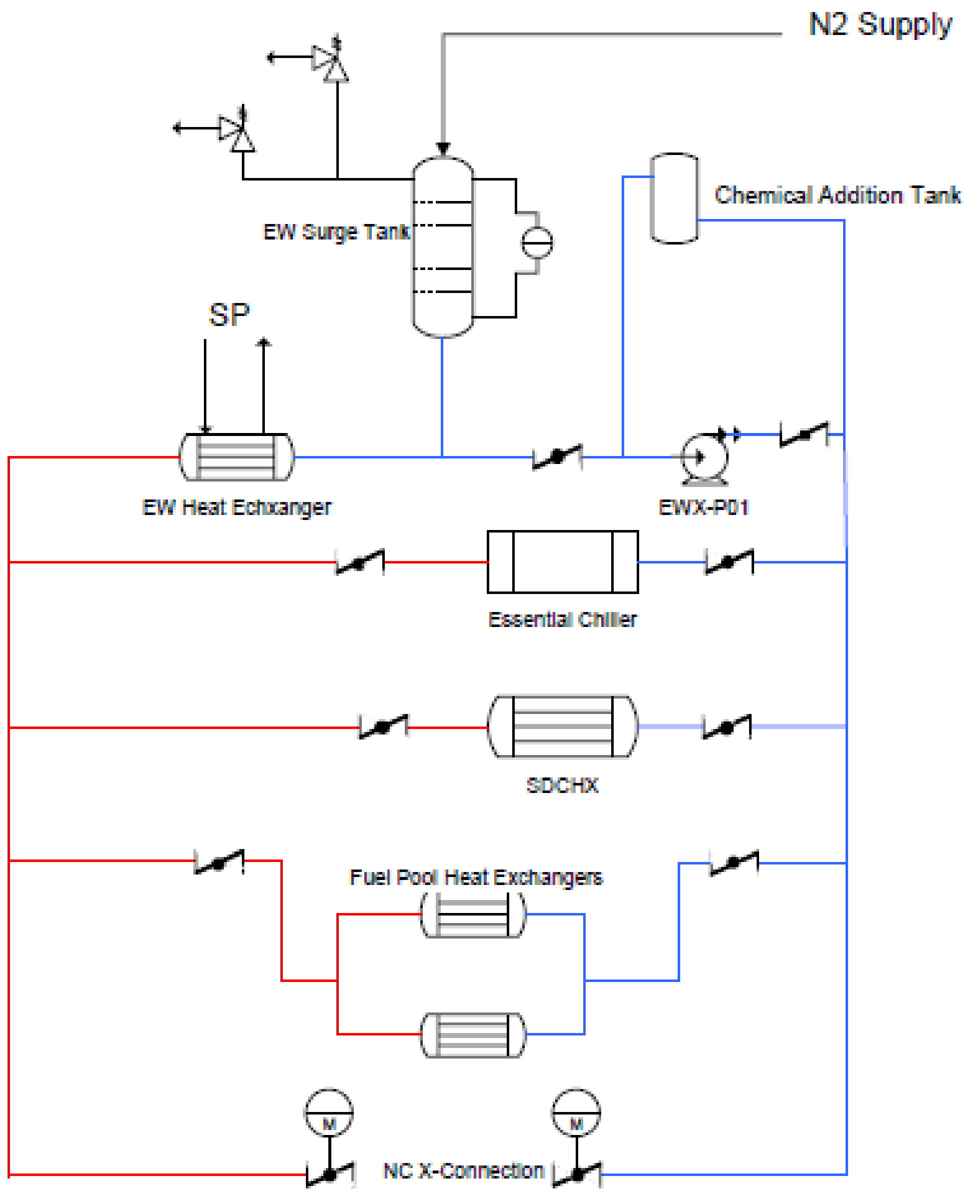
Question Source:	x	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		**Memory or Fundamental Knowledge**
	x	**Comprehension or Analysis**

Level of Difficultly:	3	
10CFR55.41:	3	
Reference Provided:	N	
Learning Objective:	**Describe the design characteristics of the Shut Down Cooling Heat Exchangers.**	

Technical Reference:	Esssential Cooling Water Tech Manual

This is a simplified drawing of the EW system. The spray pond does cool the EW system (and the RCS ultimately), however a leak in the SDCHX will go to the EW surge tank, not the spray pond.



Technical Reference:	40EP-9EO09, Functional Recovery

1.0 ENTRY CONDITIONS

1. An Extended Loss Of All AC Power (ELAP) is **NOT** in progress

   and

2. The Standard Post Trip Actions have been performed.

   or

   BOTH of the following conditions exist:

   - Event initiated from Mode 3 or Mode 4
   - LTOP is **NOT** in service

Technical Reference:	40EP-9EO11, Lower Mode Functional Recovery

1.0 ENTRY CONDITIONS

The Lower Mode Functional Recovery Procedure may be entered when **ALL** of the following conditions exist:

1. The unit is in Mode 4, 5, or 6 with LTOP in service.

   and

2. An Emergency Operating Procedure is **NOT** currently in use.

   and

3. **ANY** of the following conditions exist:
   - The CRS directs entering the LMFR
   - Any Lower Mode Safety Function Status Check Acceptance Criteria are **NOT** met
   - An Abnormal Operating Procedure directs entering the LMFR
   - An Alarm Response Procedure directs entering the LMFR
   - Any condition, or pattern of symptoms that are not being mitigated by the procedure(s) in use (Abnormal, Alarm or Normal)
   - Any condition, or pattern of symptoms for which no procedural guidance can be identified

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Emergency Core Cooling: Knowledge of ECCS design feature(s) and/or interlock(s) which provide for the following: Recirculation of minimum flow through pumps	Tier	2		
	Group	1		
	K/A	006 K4.06		
	IR	2.7		

Question 31

(1) Per 73ST-9SI11, Low Pressure Safety Injection Pumps Miniflow – Inservice Test, the MAXIMUM amount of time a LPSI Pump can be run on miniflow recirculation is…

(2) In the event of a valid RAS actuation, the LPSI Pump miniflow valves…

A. (1) 30 minutes
   (2) will automatically close

B. (1) 30 minutes
   (2) must be manually closed

C. (1) 60 minutes
   (2) will automatically close

D. (1) 60 minutes
   (2) must be manually closed

Proposed Answer:	**C**

Explanations:	
A.	First part is plausible since there are pumps which will be damaged due to overheating in 30 minutes (like RCPs on a loss of cooling), however LPSI Pumps may run for up to 60 minutes on miniflow prior to damage occurring. Second part is correct.
B.	First part is plausible since there are pumps which will be damaged due to overheating in 30 minutes (like RCPs on a loss of cooling), however LPSI Pumps may run for up to 60 minutes on miniflow prior to damage occurring. Second part is plausible since the RWT outlet valves must be manually closed following a RAS, however the LPSI Pump miniflow valves will automatically close.
C.	Correct.
D.	First part is correct. Second part is plausible since the RWT outlet valves must be manually closed following a RAS, however the LPSI Pump miniflow valves will automatically close.

Question Source:	x	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	x	**Memory or Fundamental Knowledge**
		Comprehension or Analysis

Level of Difficultly:	**3**	
10CFR55.41:	**8**	
Reference Provided:	**N**	
Learning Objective:	**Describe what will automatically initiate a Recirculation Actuation Signal (RAS) and its function.**	

Technical Reference:	73ST-9SI11, LPSI Pumps Miniflow - IST

3.2 Limitations

3.2.1 RWT level must be greater than 5% at all times during this test.

3.2.2 Pump operations shall NOT exceed 1 hour on miniflow recirculation.

3.2.3 Frequent starting may result in serious damage to the motor on a Low Pressure Safety Injection (LPSI) Pump. Anytime the motor windings are energized constitutes a start. All of the following limitations apply:

Technical Reference:	40AO-9ZZ11, Inadvertent PPS-ESFAS Actuations

Attachment C-11 RAS Train A Page 1 of 1

Actuation Leg		Component	Handswitch	Actuated Condition	In Actuated Condition (Circle one)	As Left Condition (Circle one)
1-3		HPSI Pump A Recirc Valve	SIA-HS-666	Closed	Y / N	Open / Closed
1-3		Containment Spray Pump A Recirc Valve	SIA-HS-664	Closed	Y / N	Open / Closed
1-3		LPSI Pump A Recirc Valve	SIA-HS-669	Closed	Y / N	Open / Closed
	2-4	LPSI Pump A	SIA-HS-3	Stopped	Y / N	Run / Stop
1-3		Train A Pumps Combined Recirc to RWT Valve	SIA-HS-660	Closed	Y / N	Open / Closed
1-3		Containment Sump to Safety Injection Train A Valve	SIA-HS-673	Open	Y / N	Open / Closed
1-3		Containment Sump to Safety Injection Train A Valve	SIA-HS-674	Open	Y / N	Open / Closed

Technical Reference:	40EP-9EO03, LOCA

On a RAS, the RWT Outlet valves must be manually closed:

* 58. IF a RAS has actuated,
THEN perform the following:

a. Ensure that both LPSI Pumps are stopped.

b. Ensure that the ESF pump suction has shifted to the containment.

(continue)

b.1 IF any ESF pump suctions can NOT be shifted to the containment sump,
THEN perform the following:

1) IF ANY HPSI Pump is running with its associated Containment suction closed,
THEN stop the affected HPSI Pump.

2) IF ANY CS Pump is running with its associated Containment suction closed,
THEN stop the affected CS Pump.

* 58. (continued)

c. Close BOTH of the following valves:

- CHA-HV-531, RWT to Train A Safety Injection Valve
- CHB-HV-530, RWT to Train B Safety Injection Valve

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Pressurizer Relief/Quench Tank: Knowledge of the operational implications of the following concepts as the apply to PRTS: Method of forming a steam bubble in the PZR	**Tier**	**2**		
	Group	**1**		
	K/A	**007 K5.02**		
	IR	**3.1**		

Question 32

Per 40OP-9ZZ23, Outage GOP, when forming a steam bubble in the Pressurizer during a Reactor startup, the Pressurizer should be vented to…

A. Containment

B. the Reactor Drain Tank

C. the Volume Control Tank

D. the Equipment Drain Tank

Proposed Answer:	B

Explanations:	
A.	Plausible since the Pressurizer can be vented to Containment, and Vent Valve HV-109 is in line with the vent path to Containment, however when drawing a bubble, the path through 109 goes to the RDT.
B.	Correct.
C.	Plausible since the VCT receives bleed off from the RCPs, however the Pressurizer vents used during the drawing of a bubble are vented to the RDT.
D.	Plausible as several relief valves in the CVCS system relief to the EDT, however when drawing a pressurizer bubble, the vents are directed to the RDT.

Question Source:		**New**	
	X	**Bank**	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	**Memory or Fundamental Knowledge**
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	3	
Reference Provided:	N	
Learning Objective:	**Explain the operation of the Pressurizer under normal operating conditions.**	

Technical Reference:	40OP-9ZZ23, Outage GOP

Appendix W - Forming a Steam Bubble in Pressurizer

___ 1. IF GAN-V063, LP Nitrogen Containment Header Isolation Valve, is closed, **THEN** open GAN-V063, LP Nitrogen Containment Header Isolation Valve.

___ 2. Open CHN-V483, Nitrogen Into RDT Isolation Valve.

Signature ______________________________ Date __________
Performer

___ 3. Perform an Independent Verification CHN-V483, Nitrogen Into RDT Isolation Valve, is open per 02DP-0ZZ01, Verification of Plant Activities.

Signature ______________________________ Date __________
Independent Verifier

___ 4. Open ALL of the following valves:

- ____ • RCB-HV-108 using RCB-HS-108, PRESSURIZER VENT VLV
- ____ • RCB-HV-109 using RCB-HS-109, PRZR VENT THROTTLE VLV
- ____ • RCB-HV-105 using RCB-HS-105, PRZR/RV HD VENT VLV

Technical Reference:	40OP-9ZZ23, Outage GOP

NOTE

___ CHN-PSV-354, Purification Filter Inlet Relief Valve, lifts at 190 psig (lift tolerance is between 179 psig and 201 psig).

CAUTION

___ RCS pressure greater than 250 psig when a Containment Spray pump is operating on Shutdown Cooling will exceed the SI piping design pressure.

___ 17. Operate Pressurizer heaters to raise Pressurizer pressure between 100 psia and 190 psia.

NOTE

___ The CORA program may be used to establish an alarm for venting the Reactor Vessel Head every 12 hours.

___ 18. Perform the following to vent the Pressurizer:

____ 18.1 Record date and time of the first venting cycle: ____________

____ 18.2 Perform the following every 12 hours to vent the Pressurizer steam space:

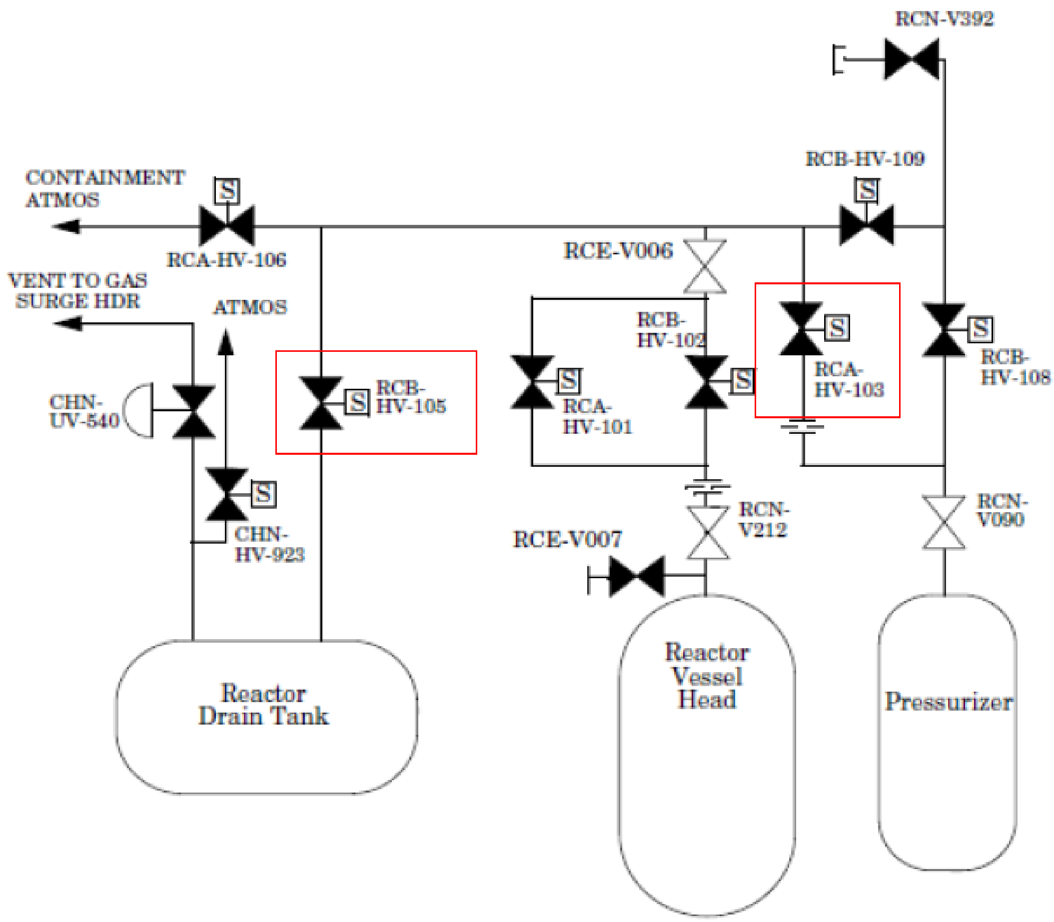
___ 18.2.1 Open RCA-HV-103 using RCA-HS-103, PRESSURIZER VENT VLV.

___ 18.2.2 Open RCB-HV-105 using RCB-HS-105, PRZR/RV HD VENT TO RDT VLV.

Technical Reference:	Reactor Coolant System Tech Manual

Venting the pressurizer can be done via HV-105 to the RDT, or via HV-106 to the Containment, however when drawing a bubble, the pressurizer is vented to the RDT

Appendix N - Reactor Vessel Head and Pressurizer Vent System



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Component Cooling Water: Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: Loads cooled by CCWS	**Tier**	**2**		
	Group	**1**		
	K/A	**008 K3.01**		
	IR	**3.4**		

Question 33

Given the following conditions:

- Unit 2 is operating at 100% power
- NCW Containment Upstream Supply Isolation Valve, NCB-UV-401, has spuriously closed and cannot be reopened

Which of the following describe the effect of this valve closure?

1. CEDM ACU outlet air temperature will rise
2. NCW temperature from the Letdown Heat Exchanger will rise
3. NCW temperature from the Nuclear Sample Coolers will rise

A. 1 ONLY

B. 2 ONLY

C. 1 and 3 ONLY

D. 2 and 3 ONLY

Proposed Answer:	A

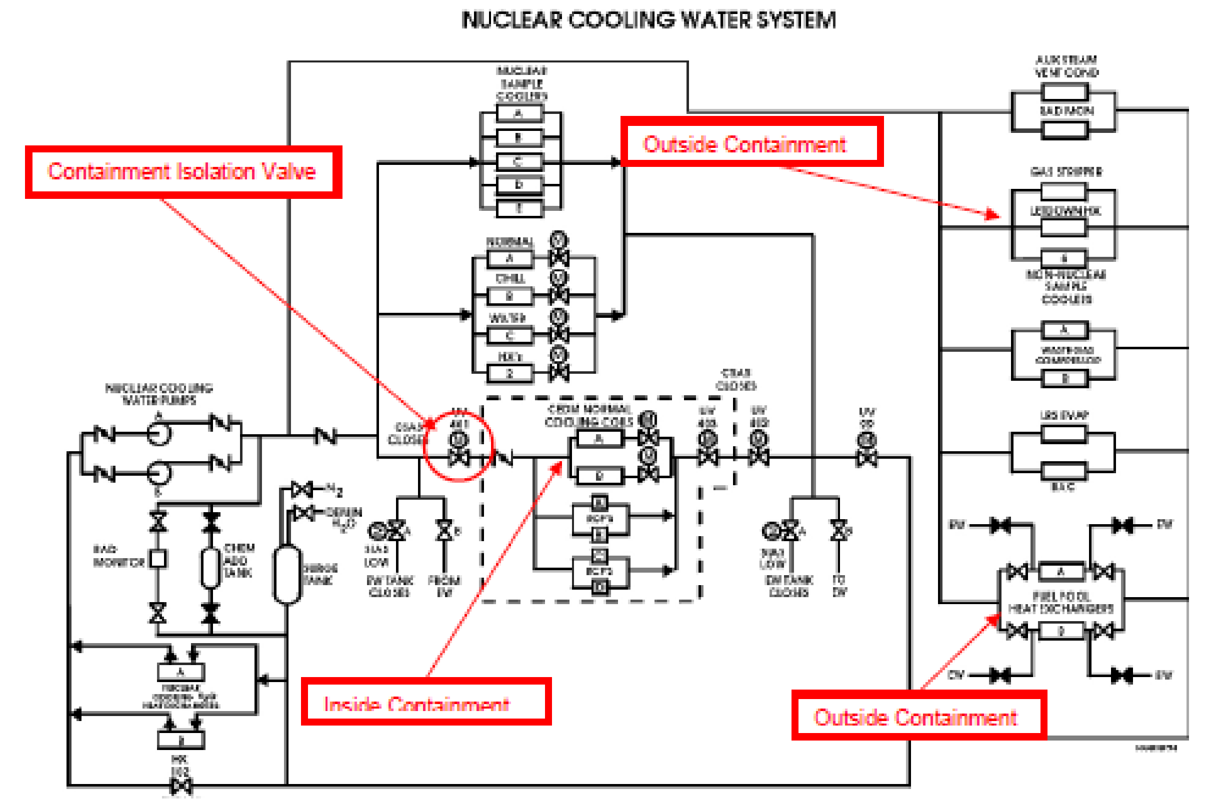
Explanations:	
A.	Correct.
B.	Plausible since the LDHX is cooled by NC and portions of the letdown and NC systems are located inside containment, however the LDHX and associated NC piping are located upstream of NCB-UV-401.
C.	CEDM ACU air temp is correct. Plausible since the NC sample coolers are a priority load cooled by NC, and all loads inside containment are priority loads, however the sample coolers are still cooled following the closure of UV-401.
D.	LDHX is plausible since the LDHX is cooled by NC and portions of the letdown and NC systems are located inside containment, however the LDHX and associated NC piping are located upstream of NCB-UV-401. Plausible since the NC sample coolers are a priority load cooled by NC, and all loads inside containment are priority loads, however the sample coolers are still cooled following the closure of UV-401.

Question Source:		New	
	x	Bank	
		Modified	
	x	Previous NRC Exam	2016 NRC Q35

Cognitive Level:		Memory or Fundamental Knowledge
	x	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	4	
Reference Provided:	N	
Learning Objective:	Describe the Control Room indications associated with the Nuclear Cooling Water system.	

Technical Reference:	Nuclear Cooling Water Tech Manual



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Pressurizer Pressure Control: Knowledge of bus power supplies to the following: Controller for PZR spray valve	**Tier**	**2**		
	Group	**1**		
	K/A	**010 K2.02**		
	IR	**2.5**		

Question 34

Which of the following is the source of power to Pressurizer Spray Controller, RCN-PIK-100?

A. Class 120 VAC power

B. Class 125 VDC power

C. Non-Class 120 VAC power

D. Non-Class 125 VDC power

Proposed Answer:	**C**

Explanations:	
A.	Plausible since PIK-100 is powered by 120 VAC, however it is non-class, not class powered.
B.	Plausible since Aux Spray Valves are powered by class 125 VDC power, however the PIK-100 is powered by non-class 120 VAC power
C.	Correct.
D.	Plausible since PIK-100 is non-class, and power to the Aux Spray Valves is DC, however PIK-100 is powered from 120 VAC non-class power

Question Source:	**x**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	**x**	**Memory or Fundamental Knowledge**
		Comprehension or Analysis

Level of Difficultly:	**3**	
10CFR55.41:	**3**	
Reference Provided:	**N**	
Learning Objective:	**Describe the various other operating systems the Non-Class IE Instrument AC System (NN) supports.**	

Technical Reference:	PPCS Tech Manual

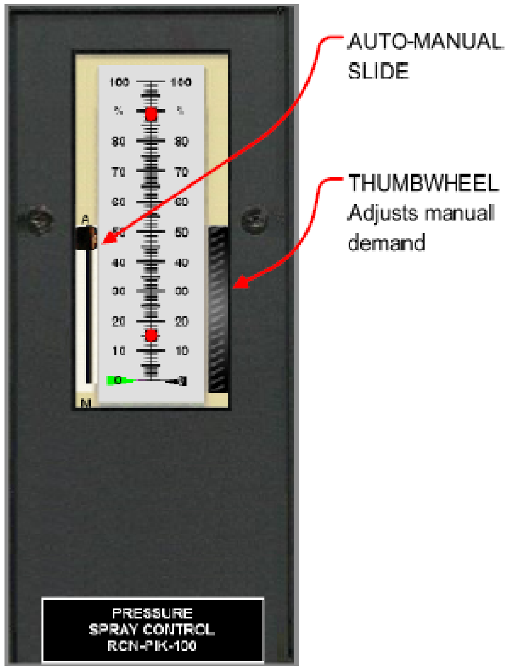
RCN-PIK-100 is powered from NNN-D12, which is a non-class 120 VAC bus

2.12 Spray Valve Controller RCN-PIK-100

RCN-PIK-100, located on B04, provides the modulation signal to position the selected spray valve(s). An "A"-"M" (auto-manual) slide switch allows the operator to select the controller mode of operation. The spray valve controller has a 0-100% vertical meter that displays dual indications. A green needle indicates the auto demand signal to the spray valves. This signal is generated by RCN-PIK-100 in response to input from RCN-PIC-100. A 33% to 50% input from RCN-PIC-100 yields a 0% to 100% auto demand signal. The black needle indicates the manual demand setting. A thumbwheel allows adjustment of the manual demand.

In automatic, the auto demand signal developed from RCN-PIC-100 input is fed to the electro-pneumatic converter(s) to position the spray valve(s). In manual, the auto demand remains indicated on the vertical meter but is blocked from the spray valves. A signal based on thumbwheel position is then fed to the electro-pneumatic converter(s). There is no indication of actual controller output in the manual mode.

RCN-PIK-100 receives power from NNN-D12. A loss of power to this controller wil result in the controller failing to zero output (0 ma). This will result in the closure of both spray valves.



RCN-PIK-100

Examination Outline Cross-Reference:	**Level**	**RO**	[redacted]	**SRO**
K/A: Reactor Protection: Knowledge of the operational implications of the following concepts as the apply to the RPS: Power Density	**Tier**	**2**	[redacted]	
	Group	**1**	[redacted]	
	K/A	**012 K5.02**		
	IR	**3.1**	[redacted]	

Question 35

Given the following conditions:

- Unit 2 is preparing to commence a Reactor Startup
- DNBR and LPD trips are bypassed at the CPCs

During the startup, the LPD trip bypasses ___(1)___ removed, and after the bypasses have been removed, the Reactor will trip if LPD rises to a MINIMUM of ___(2)___ .

A. (1) must be manually
   (2) 13.1 kw/ft

B. (1) must be manually
   (2) 21 kw/ft

C. (1) will be automatically
   (2) 13.1 kw/ft

D. (1) will be automatically
   (2) 21 kw/ft

Proposed Answer:	D

Explanations:	
A.	First part is plausible since the LPD trips must be manually bypassed, however they will automatically come out of bypass as power rises. Second part is plausible since 13.1 kw/ft is the LHR limit in the COLR, however the LPD trip is 21 kw/ft.
B.	First part is plausible since the LPD trips must be manually bypassed, however they will automatically come out of bypass as power rises. Second part is correct.
C.	First part is correct. Second part is plausible since 13.1 kw/ft is the LHR limit in the COLR, however the LPD trip is 21 kw/ft.
D.	Correct.

Question Source:	x	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	x	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	7	
Reference Provided:	N	
Learning Objective:	List the parameters and setpoints that will cause PPS actuation.	

Technical Reference:	LOIT Plant Protection System Lesson Plan

DNBR/LPD Bypass

The DNBR/LPD bypass defeats both the DNBR and LPD trips from the CPCs. It allows a normal reactor startup, since an abnormal CEA configuration, such as shutdown CEAs inserted, will cause DNBR and LPD trips. Each protection channel must be bypassed individually. All four channels may be bypassed simultaneously. In accordance with 40OP-9ZZ03 (Outage GOP) the bypass must be manually inserted from key switches at the remote CPC modules on B05 when ex-core safety channel NI power is less than 1x10-5% power. The bypass will be automatically removed should ex-core NI power increase above 10-4%. It may also be manually removed.

Technical Reference:	B05A Alarm Response Procedure

Panel B05A Alarm Responses	40AL-9RK5A	Revision 2
	Page 1 of 5	

Response Section

5A14C

HI LPD CH TRIP

High Local Power Density Channel Trip

Point ID	Description	Setpoint
SBTA03	Hi Local Power Density Channel A Trip	21.0 kw/ft
SBTB03	Hi Local Power Density Channel B Trip	21.0 kw/ft
SBTC03	Hi Local Power Density Channel C Trip	21.0 kw/ft
SBTD03	Hi Local Power Density Channel D Trip	21.0 kw/ft

Technical Reference:	PVNGS Core Operating Limits Report

3.2.1 - Linear Heat Rate (LHR)

The linear heat rate limit of 13.1 kW/ft shall be maintained.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Engineered Safety Features Actuation: Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations; LOCA	Tier	2		
	Group	1		
	K/A	013 A2.01		
	IR	4.6		

Question 36

Given the following conditions:

- Unit 3 is operating at 2% power
- Both SGs are being fed from AFN-P01

Subsequently:

- The Reactor was tripped due to a LOCA
- Pressurizer level is 15% and lowering
- Pressurizer pressure is 1800 psia and lowering
- SG levels are both 20% NR and slowly lowering
- SG pressures are 1150 psia and slowly lowering
- Containment pressure is 2.0 psig and slowly rising

With NO operator action, which of the following describes the current status of the Auxiliary Feedwater Pumps and feed to the SGs?

Auxiliary Feedwater Pump(s) ___(1)___ is(are) running and the SGs are ___(2)___ .

A. (1) AFB-P01 ONLY
   (2) being fed

B. (1) AFB-P01 ONLY
   (2) NOT being fed

C. (1) AFN-P01 AND AFB-P01
   (2) being fed

D. (1) AFN-P01 AND AFB-P01
   (2) NOT being fed

Proposed Answer:	B

Explanations:	
A.	First part is correct (AFN-P01 trips on SIAS). Second part is plausible since the SGs were being fed before the LOCA and there is still an AFW Pump running, however the feed paths are different and no actuation at this point would reinitiate feed to the SGs.
B.	Correct.
C.	First part is plausible since AFN-P01 was running and AFB-P01 starts on the SIAS actuation (actuates at 1837 psia – RCS pressure), however the SIAS actuation also trips AFN-P01. Second part is plausible since the SGs were being fed before the LOCA and there is still an AFW Pump running, however the feed paths are different and no actuation at this point would reinitiate feed to the SGs.
D.	First part is plausible since AFN-P01 was running and AFB-P01 starts on the SIAS actuation (actuates at 1837 psia – RCS pressure), however the SIAS actuation also trips AFN-P01. Second part is correct.

Question Source:	x	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	x	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	7	
Reference Provided:	N	
Learning Objective:	Describe the automatic functions / interlocks associated with AFN-P01.	

Technical Reference:	40AO-9ZZ17, Inadvertent PPS-ESFAS Actuations

The SIAS, which actuates when RCS pressure is < 1837 psia, will trip AFN-P01 and start AFB-P01

Attachment C-13 SIAS Train A Page 3 of 4

Actuation Leg		Component	Handswitch	Actuated Condition	In Actuated Condition (Circle one)	As Left Condition (Circle one)
1-3		SIT 1A Outlet to RC Loop 1A Valve	SIA-HS-634	Open	Y / N	Open / Closed
1-3		SIT 1B Outlet to RC Loop 1B Valve	SIA-HS-644	Open	Y / N	Open / Closed
1-3		Misc Drain Header To RWT Valve	SIA-HS-682	Closed	Y / N	Open / Closed
1-3		Letdown to Regen Hx Isolation Valve	CHA-HS-516	Closed	Y / N	Open / Closed
	2-4	Backup Heaters Bank	RCA-HS-100-4	Tripped	Y / N	Tripped / Closed
	2-4	S/U Aux Feed Pump	AFA-HS-11	Stopped	Y / N	Run / Stop
	2-4	Condensate Transfer Pump A	CTA-HS-15	Running	Y / N	Run / Stop

Attachment C-14 SIAS Train B Page 3 of 3

1-3		SI Line to RC Loop 1A Drain Valve	SIB-HS-638	Closed	Y / N	Open / Closed
1-3		SI Line to RC Loop 1B Drain Valve	SIB-HS-648	Closed	Y / N	Open / Closed
1-3		SI Line to RC Loop 2A Drain Valve	SIB-HS-618	Closed	Y / N	Open / Closed
1-3		SI Line to RC Loop 2B Drain Valve	SIB-HS-628	Closed	Y / N	Open / Closed
1-3		Letdown To Regen Hx Isolation Valve	CHB-HS-515	Closed	Y / N	Open / Closed
	2-4	Backup Heaters Bank	RCB-HS-100-5	Tripped	Y / N	Tripped / Closed
	2-4	Condensate Transfer Pump B	CTB-HS-16	Running	Y / N	Run / Stop
	2-4	Essential Electric Auxiliary Feed Pump	AFB-HS-10	Running	Y / N	Run / Stop
	2-4	Containment Normal ACU Fan B	HCB-HS-12	Stopped	Y / N	Run / Stop

Technical Reference:	

Although an Aux Feed Pump is running, the feed valves have no signal to open. If AFAS had actuated at this point, which is plausible since it actuates when SG levels lower to 25.8%, however that is WR, not NR

AFW Regulating and Isolating Valve Controls

These regulating valves can be operated by separate switches on panel B06 and also at the remote shutdown panels (figure A-3). The flow regulating globe valves are equipped with JOG OPEN/JOG CLOSED handswitches. The valves may be throttled assuming an AFAS signal is not present which fully opens the valve.

The isolation gate valves have two position OPEN/CLOSE handswitches, and are also opened on the associated AFAS. An override pushbutton for each valve is provided on B06 and at the remote shutdown panels. When depressed, with an AFAS present, the valve is placed in an "override" condition. This allows the valves to be repositioned with an AFAS signal present. When override has been selected, automatic system response for steam generator level control is prevented.

An AFAS 1 or 2 signal will provide full AFW flow to the associated steam generator when its water level decreases to 25.8% WR (wide range) on two out of four class WR level instruments. When generator level increases 40.8% WR, the low level signal is removed and the valves will close. They will not reverse direction during travel unless the AFAS level setpoint (25.8% WR) is again reached. The valves will automatically cycle on the SG level signal until placed in override or the AFAS initiation signal is cleared and reset. The position of the valves can then be changed by the operator.

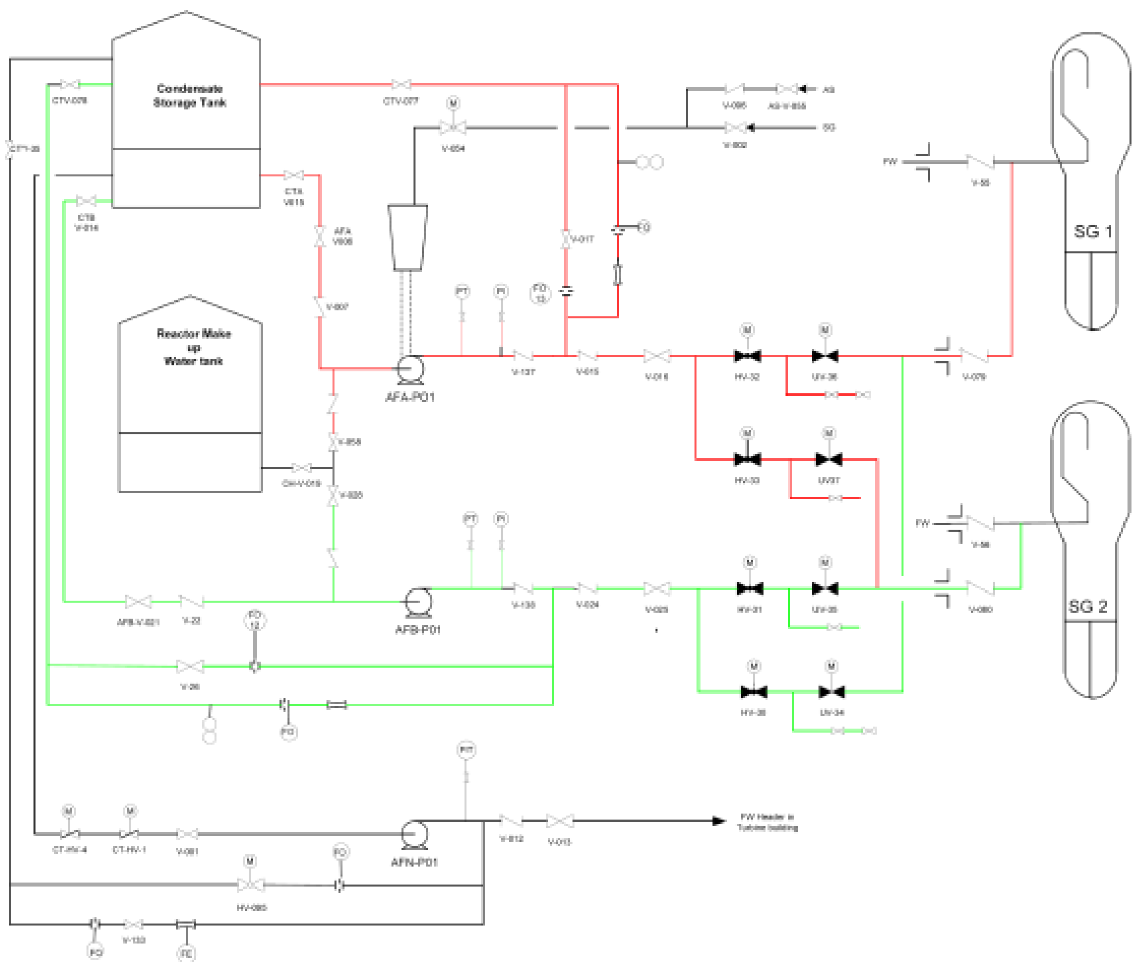
Limit switches stop valve motion in the open direction and torque protection stops closing motion when the predetermined torque value has been reached.

If an AFAS-1 signal is activated (SG #1 low level), UV-30, UV-32 and UV-34, UV-36 will all open to align AFW flow to the #1 generator. The valves will then cycle as previously described.

If an AFAS-2 signal is activated (SG #2 low level), UV-31, UV-33 and UV-35, UV-37 will all open to align AFW flow to the #2 generator. The valves will then cycle as previously described.

Technical Reference:	

Additionally, even though the SGs were being fed from AFN, and AFB is now running, they feed via different pathways so the SGs won't continue being fed even though there is still a running AFW Pump



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Containment Cooling: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.	**Tier**	**2**		
	Group	**1**		
	K/A	**022 G 2.1.31**		
	IR	**4.6**		

Question 37

Given the following conditions:

- Unit 1 is operating at 100% power
- The ‘A’ and ‘C’ Containment Normal ACU Fans are running
- The ‘B’ and ‘D’ Containment Normal ACU Fans are in standby

Subsequently:

- An inadvertent Train ‘A’ SIAS occurred
- 5 minutes after the Train ‘A’ SIAS, the CRS directs one of the ROs to verify the status of the Containment Normal ACU Fans

The RO will verify the status of the Containment Normal ACU Fans on ___(1)___ and should expect to see a total of ___(2)___ Containment Normal ACU fans running.

A. (1) Board 2
   (2) 2

B. (1) Board 2
   (2) 4

C. (1) Board 7
   (2) 2

D. (1) Board 7
   (2) 4

Proposed Answer:	C

Explanations:	
A.	First part is plausible since Control Room and Aux Building HVAC components are located on Board 2 (as well as all SI pumps and valves), however Containment HVAC components are on Board 7. Second part is correct.
B.	First part is plausible since Control Room and Aux Building HVAC components are located on Board 2 (as well as all SI pumps and valves), however Containment HVAC components are on Board 7. Second part is plausible since a SIAS is often generated due to a high energy break inside containment and it would be desired to have additional containment cooling in service, however on a SIAS, containment ACUs trip and containment cooling is provided by containment spray (if CSAS actuates).
C.	Correct.
D.	First part is correct. Second part is plausible since a SIAS is often generated due to a high energy break inside containment and it would be desired to have additional containment cooling in service, however on a SIAS, containment

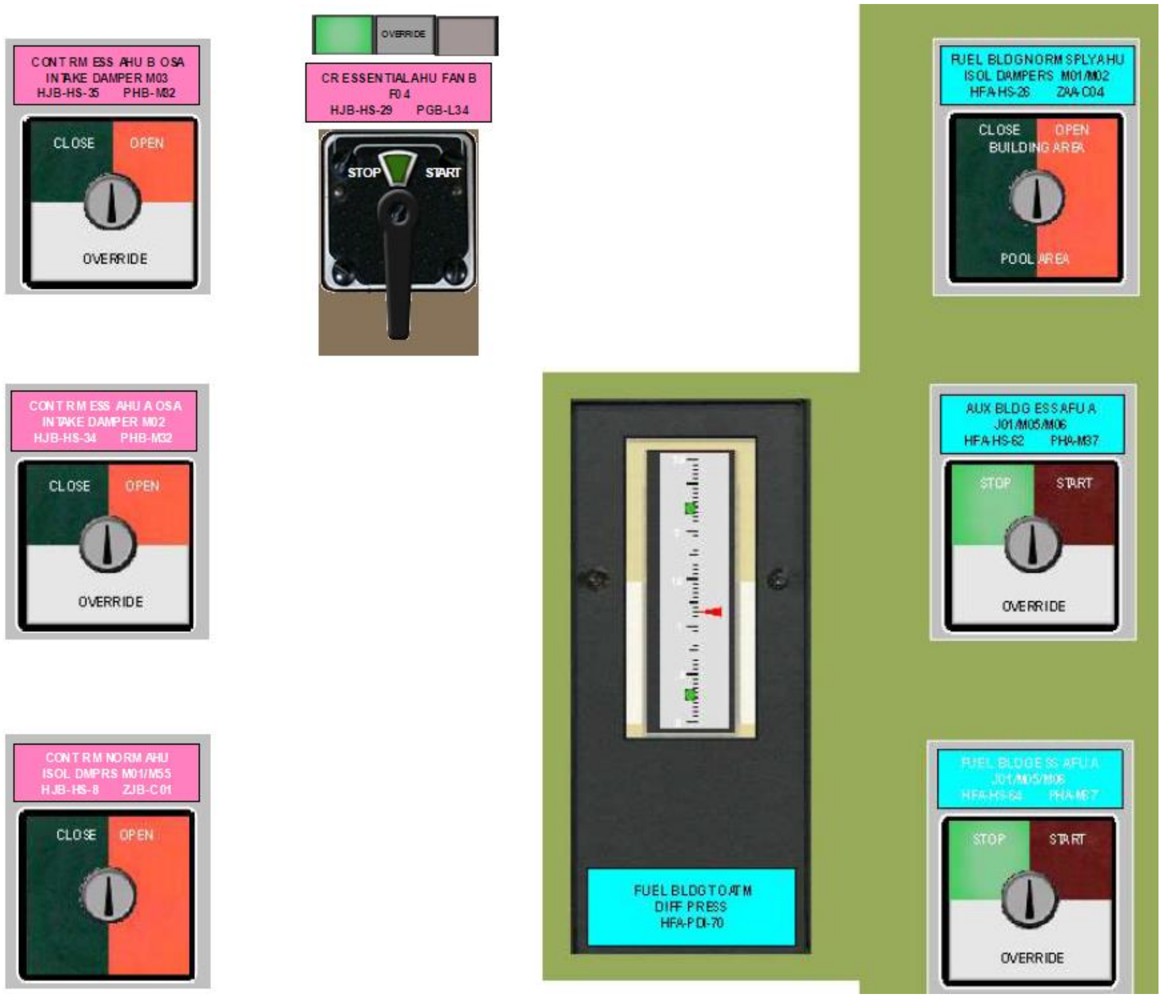
Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	7	
Reference Provided:	N	
Learning Objective:	Describe the automatic functions associated with the Containment Building Normal ACU Fans (HCNA01A, B, C, D).	

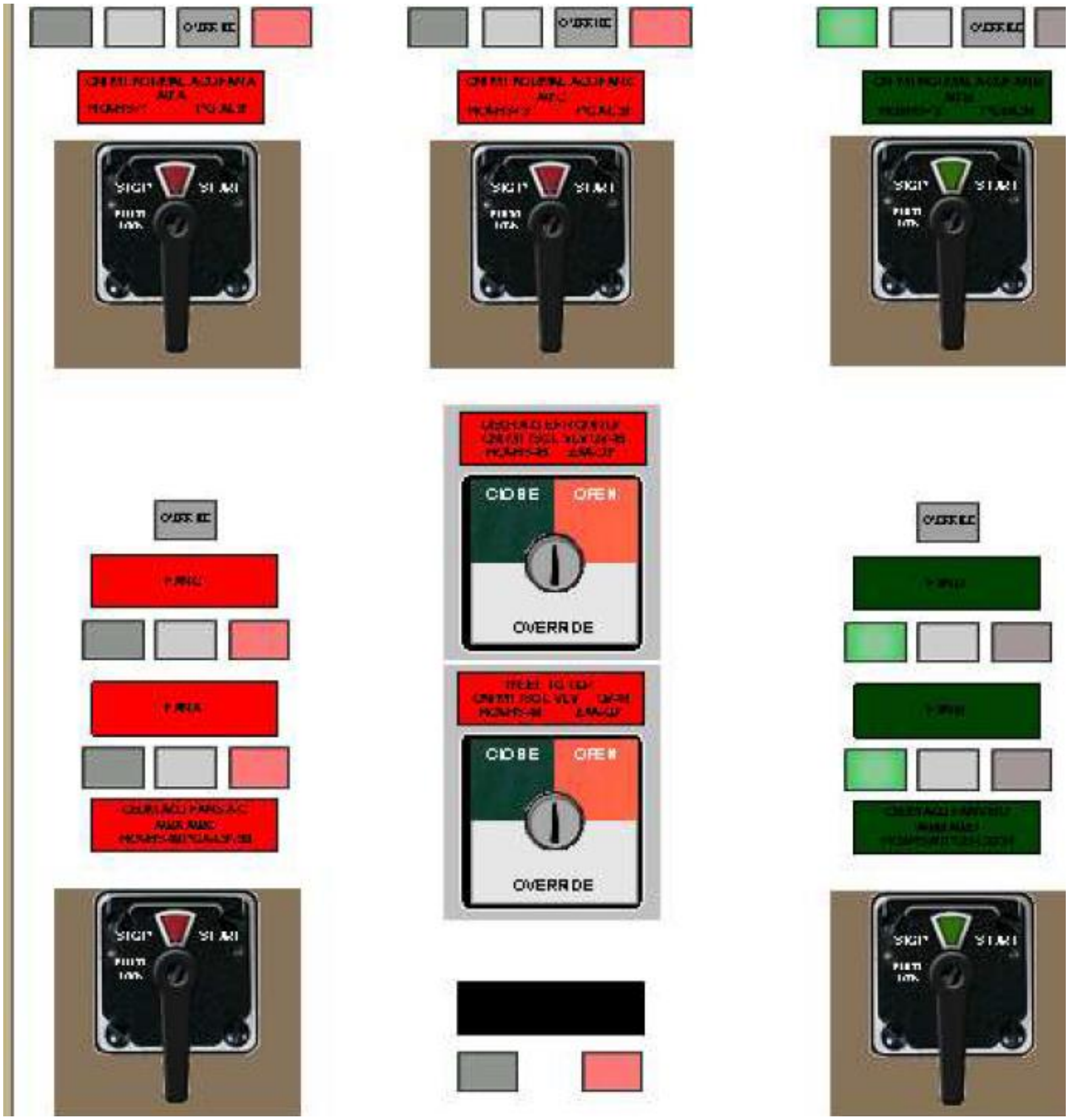
Technical Reference:	Screen Shot of B02

In this shot are the Control Room, Fuel Building, and Aux Building Air Handling Unit (ACU) controls – all on Board 2



Technical Reference:	Screen Shot of B07

In this shot, although illegible in the picture we have, are the two Containment Normal AHU control switches (two red-flagged handswitches at the top) and the CEDM ACU handswitch (red-flagged at the bottom left of the pic)



Technical Reference:	Containment Building HVAC Tech Manual

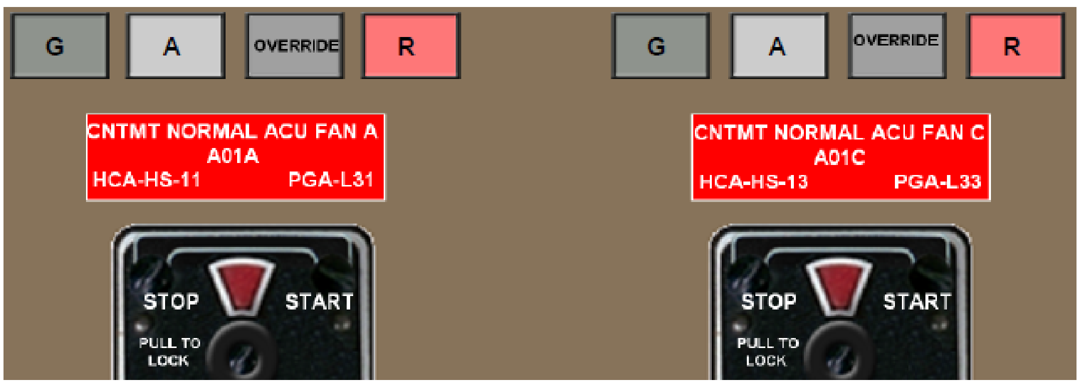
The 'A' SIAS will trip the A/C Containment Normal ACUs, B/D will start on low diff pressure, and A/C won't auto restart after the load shed sequence is complete

Normal Cooling System ACU Controls (HS-11, 12, 13, and 14)

A four position (START/STOP/PULL TO LOCK/AUTO) spring return to AUTO control switch is provided on B07 in the Control Room for each ACU. When momentarily placed in the START position, the associated breaker closes starting the ACU fan and the associated discharge damper and cooling water supply valve opens. When momentarily placed in the STOP position, the associated breaker opens, stopping the fan and closing the discharge damper and cooling water supply valve. The switches spring return to the AUTO position. When in the PULL TO LOCK position, the ACU control circuit is de-energized, preventing operation.

When in the AUTO (after stop) position, the standby ACU automatically starts if low differential pressure is sensed on the associated operating fan. ACUs in AUTO (after stop) will automatically start in accordance with diesel generator load sequencing following a loss of off-site power. Operating ACUs will trip upon receipt of a Safety Injection Actuation Signal (SIAS) or load shed signal. Following receipt of a SIAS, the operator can return the ACU to service by momentarily positioning the control switch to STOP and then to START. When moved to STOP, a white OVERRIDE indicator illuminates indicating the trip can be overridden. When placed in override the ACU will start and will continue to run even if the SIAS is reset.

The Normal ACU is equipped with a SIAS stop interlock which prevents it from auto starting when the SIAS is reset. Upon the receipt of a SIAS, the interlock causes a latching relay to change state, blocking the ACUs auto start feature. In order to restore the auto start feature, the hand switch must be taken to stop. This action energizes the latching relay, returning it to the pre-SIAS condition.



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Containment Spray: Knowledge of bus power supplies to the following: MOVs	Tier	2		
	Group	1		
	K/A	026 K2.02		
	IR	3.4		

Question 38

Containment Spray Discharge to Spray Header Valve, SIB-UV-671, is powered by…

A. PGA-L31

B. PGB-L32

C. PHA-M35

D. PHB-M36

Proposed Answer:	**D**

Explanations:	
A.	Plausible since UV-671 is powered by 480 VAC class power, and generally speaking, odd numbered valves are powered by Train ‘A’, however UV-671 is powered from PHB-M36
B.	Plausible since UV-671 is powered by 480 VAC class power, and is powered from Train ‘B’, however UV-671 is powered from PHB-M36
C.	Plausible since UV-671 is powered by 480 VAC class power, and generally speaking, odd numbered valves are powered by Train ‘A’, however UV-671 is powered from PHB-M36
D.	Correct.

Question Source:	**x**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	**x**	**Memory or Fundamental Knowledge**
		Comprehension or Analysis

Level of Difficultly:	**2**	
10CFR55.41:	**6**	
Reference Provided:	**N**	
Learning Objective:	**Describe how the Class IE Electrical Distribution System supports the operation of the following systems:** **• Safety Injection and Shutdown Cooling System**	

Technical Reference:	Safety Injection Tech Manual

Xfmr	
LPSI B Flow Control to RC2B Vlv SIB-UV-625	PHB-M3621
SIT 2 B Isol Vlv SIB-UV-624	PHB-M3618
SIT 2A Isol Vlv SIB-UV-614	PHB-M3619
Ind. Lights and Space Heater for Valve SIB-UV-614	PHB-D3601
Ind. Lights and Space Heater for Valve SIB-UV-264	PHB-D3602
Ctmt Sump Isol Train B Vlv SIB-UV-676	PHB-M3614
Ctmt Spray Isol Train B Valve SIB-UV-671	PHB-M3612
Ctmt Sump Isol Train B Vlv SIB-UV-675	PHB-M3613
HPSI Pump B to RWT Isol Vlv SIB-UV-667	PHB-M3608
LPSI Flow Control to RC2A Vlv SIB-UV-615	PHB-M3606
SDC Control Isol Loop B Vlv SIB-UV-656	PHB-M3605
Ctmt Spray Pump B to RWT Isol Vlv SIB-UV-665	PHB-M3607
HPSI B Flow Control to RC 1B Vlv SIB-UV-646	PHB-M3603
SDC Isol Loop B Vlv SIB-UV-652	PHB-M3604

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Main and Reheat Steam: Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following: Automatic isolation of steam line	**Tier**	**2**		
	Group	**1**		
	K/A	**039 K4.05**		
	IR	**3.7**		

Question 39

A Main Steam Isolation Signal will actuate on ___(1)___ SG level, and if the condition in Part 1 occurs in SG #1 ONLY, the MSIVs will close on ___(2)___ .

A. (1) low
   (2) SG #1 ONLY

B. (1) low
   (2) BOTH SGs

C. (1) high
   (2) SG #1 ONLY

D. (1) high
   (2) BOTH SGs

Proposed Answer:	D

Explanations:	
A.	First part is plausible since low SG level is indicative of either insufficient feed or a steam line break, and in either case, stopping steam flow would be desired, however an MSIS is automatically actuated due to low SG pressure in those cases and only actuates due to level when it is high. Second part is plausible since most ESFAS actuations are train specific (i.e. SIAS, CSAS), MSIS is selectively SG specific (i.e. SG blowdown valves), however MSIVs are all closed on both SGs regardless of which SG reached the actuation setpoint.
B.	First part is plausible since low SG level is indicative of either insufficient feed or a steam line break, and in either case, stopping steam flow would be desired, however an MSIS is automatically actuated due to low SG pressure in those cases and only actuates due to level when it is high. Second part is correct.
C.	First part is correct. Second part is plausible since most ESFAS actuations are train specific (i.e. SIAS, CSAS), MSIS is selectively SG specific (i.e. SG blowdown valves), however MSIVs are all closed on both SGs regardless of which SG reached the actuation setpoint.
D.	Correct.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	**X**	**Memory or Fundamental Knowledge**
		Comprehension or Analysis

Level of Difficultly:	**2**	
10CFR55.41:	**7**	
Reference Provided:	**N**	
Learning Objective:	**Describe the automatic functions associated with the Main Steam System.**	

Technical Reference:	Plant Protection System Tech Manual

- Main steam (SG)

Each steam generator provides two water level signals to the PPS to be used as PPS bistable inputs. The SG wide range (WR) signals are used for the low steam generator water level reactor trip and auxiliary feedwater actuation system (AFAS) actuations. The SG narrow range (NR) signals are used for the high steam generator water level reactor trip and main steam isolation system (MSIS) actuations.

Pressure signals from each SG are also provided as inputs. These signals are used for the low steam generator pressure reactor trip, AFAS SG rupture logic and main steam isolation system (MSIS) actuations.

Technical Reference:	40AO-9ZZ17, Inadvertent PPS-ESFAS Actuations

MSIV 170 and 180 are the Line 1 and Line 2 MSIVs for SG1, MSIV 171 and 181 are the Line 1 and Line 2 MSIVs for SG2. MSIS doesn't only close MSIVs on the SG which sensed the low pressure or high level, but it closes all MSIVs regardless of which SG actuated it.

Attachment C-10 MSIS Train B Page 1 of 2

Actuation Leg		Component	Handswitch	Actuated Condition	In Actuated Condition (Circle one)	As Left Condition (Circle one)
	2-4	SG 1 Chemical Injection	SGB-HS-200	Closed	Y / N	Open / Closed
1-3		MSIV Bypass Isolation Valve	SGB-HS-169B	Closed	Y / N	Open / Closed
1-3		Line 1 MSIV	SGB-HS-170B	Closed	Y / N	Open / Closed
1-3		Line 2 MSIV	SGB-HS-180B	Closed	Y / N	Open / Closed
1-3		Economizer FWIV	SGB-HS-132A	Closed	Y / N	Open / Closed
1-3		Downcomer Isolation Valve	SGB-HS-130	Closed	Y / N	Open / Closed
1-3		Economizer FWIV	SGB-HS-137A	Closed	Y / N	Open / Closed
1-3		Downcomer Isolation Valve	SGB-HS-135	Closed	Y / N	Open / Closed
1-3		MSIV Bypass isolation Valve	SGB-HS-183B	Closed	Y / N	Open / Closed
1-3		Line 1 MSIV	SGB-HS-171B	Closed	Y / N	Open / Closed
1-3		Line 2 MSIV	SGB-HS-181B	Closed	Y / N	Open / Closed

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Main Feedwater: Ability to monitor automatic operation of the MFW, including: Programmed levels of the S/G	**Tier**	**2**		
	Group	**1**		
	K/A	**059 A3.02**		
	IR	**2.9**		

Question 40

During a power ascension, DFWCS swapover will occur when selected power is > 15% and ___(1)___ of the downcomer valves is(are) at least 80% open, OR when selected power reaches a MINIMUM of ___(2)___ , regardless of downcomer valve positions.

A. (1) BOTH
(2) 16.5%

B. (1) BOTH
(2) 18%

C. (1) EITHER
(2) 16.5%

D. (1) EITHER
(2) 18%

Proposed Answer:	**C**

Explanations:	
A.	First part is plausible since it makes sense that both downcomer valves would need to be > 80% open in order for swapover to occur since maintaining SG levels approximately equal is desired for balanced heat removal, however at greater than 15% power, swapover will occur when the first valve is 80% open. Second part is correct.
B.	First part is plausible since it makes sense that both downcomer valves would need to be > 80% open in order for swapover to occur since maintaining SG levels approximately equal is desired for balanced heat removal, however at greater than 15% power, swapover will occur when the first valve is 80% open. Second part is plausible since at 18% power, the downcomer valves should be ~ 85% open (based on downcomer program), however at 16.5% power swapover will occur regardless of downcomer valve position.
C.	Correct.
D.	First part is correct. Second part is plausible since at 18% power, the downcomer valves should be ~ 85% open (based on downcomer program), however at 16.5% power swapover will occur regardless of downcomer valve position.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		**Memory or Fundamental Knowledge**
	x	**Comprehension or Analysis**

Level of Difficultly:	**3**	
10CFR55.41:	**4**	
Reference Provided:	**N**	
Learning Objective:	**Describe the response of the DFWCS to an increase in reactor power to include the following: Swapover**	

Technical Reference:	LOIT DFWCS Lesson Plan

The control channels are used for:

- System response adjustment: The DFWCS uses reactor power as one of the inputs that will tune the responsiveness of the system by adjusting the proportional band and integration rate based on the reactor power level. The system is more responsive at higher power levels.

- Changing total feedwater flow indication from downcomer flow to total feedwater flow at 13% and increasing reactor power.

- Transition from single element control to 3 element control: at 14% and increasing reactor power the DFWCS transitions from single element control to 3 element control.

- Transition from 3 element control to single element control: at below 13.5% and decreasing reactor power the DFWCS transitions from 3 element control to single element control.

- Swapover: the shifting of feedwater flow from through the Downcomer control valves to through the Economizer control valves (during a increase in reactor power) or from through the Economizer control valves to through the Downcomer control valves (during a decrease in reactor power) is known as swapover. Swapover during an increase in reactor power will occur if reactor power is above 16.5% or if reactor power is between 15 and 16.5% and either Downcomer control valve reaches 80% open. Swapover during a decrease in reactor power will occur if reactor power is below 15%.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Auxiliary/Emergency Feedwater: Knowledge of bus power supplies to the following: AFW system MOVs	Tier	2		
	Group	1		
	K/A	061 K2.01		
	IR	4.0		

Question 41

(1) AFW Regulating Valve from AFA-P01 to SG1, AFA-HV-32, is powered from…

(2) AFW Regulating Valve from AFB-P01 to SG1, AFB-UV-30, is powered from…

A. (1) Class 125 VDC power
   (2) Class 125 VDC power

B. (1) Class 125 VDC power
   (2) Class 480 VAC power

C. (1) Class 480 VAC power
   (2) Class 125 VDC power

D. (1) Class 480 VAC power
   (2) Class 480 VAC power

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible as some AF MOVs are DC powered, however the Train B AFW MOVs are 480 VAC powered.
B.	Correct.
C.	First part is plausible as some AF MOVs are AC powered, however the Train A AFW MOVs are 125VDC powered. Second part is plausible as some AF MOVs are DC powered, however the Train B AFW MOVs are 480 VAC powered.
D.	First part is plausible as some AF MOVs are AC powered, however the Train A AFW MOVs are 125VDC powered. Second part is correct.

Question Source:		**New**	
	X	**Bank**	
		Modified	
	x	**Previous NRC Exam**	**2018 NRC Q44**

Cognitive Level:	**X**	**Memory or Fundamental Knowledge**
		Comprehension or Analysis

Level of Difficultly:	**3**	
10CFR55.41:	**4**	
Reference Provided:	**N**	
Learning Objective:	**Describe how the AF System is supported by the following systems:** • **Class 1E 125 VDC System** • **Class 1E 480 VAC System**	

Technical Reference:	Aux Feedwater System Tech Manual

PKA is a 125 VDC Control Power Bus, and PHB is a 480VAC Loadcenter

Table C - 1

AF System Electrical Power Supplies

COMPONENT	POWER SUPPLY
Aux. Feedwater Pump B	PBB-S04S
MOV POSIT Indicators at Aux Relay Cab ZAA-C01 (AFC-UV-32 Psn Indication)	PNA-D2517
MOV POSIT Ind at Aux Relay Cab ZAA-C01 (AFA-UV-33 Psn Indication)	PNA-D2519
BOP Analog Inst Cabinet & Indicators ZJA-C02A & B (Aux FW to SG1 AF FI-40)	PNA-D2524
Distribution Panel PKA-4121	PKA-M4121
Auxiliary Feedwater Turbine Gov Control	PKA-D2118
Auxiliary Feedwater Reg Valve AFA-HV-32	PKA-M4112
Auxiliary Feedwater Isolation Valve AFA-UV-37	PKA-M4113
Auxiliary Feedwater Turbine Trip & Throttle Valve AFA-HV-54	PKA-M4114
BOP Analog Instr Cabinet & Indicators ZJB-C02A (Aux FW to SG2 AF FI-41)	PNB-D2624
Auxiliary Feedwater Isolation Valve AFC-UV-36	PKC-M4314
Auxiliary Feedwater Reg Valve AFC-UV-33	PKC-M4315
Aux Feedwater Reg Valve Pump B to SG1 AFB-UV-30	PHB-M3420
Aux Feedwater Reg Valve Pump B to SG2 AFB-UV-31	PHB-M3421
Aux Feedwater Isol Valve Pump B to SG1 AFB-UV-34	PHB-M3814
Aux Feedwater Isol Valve Pump B to SG2 AFB-UV-35	PHB-M3815

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: AC Electrical Distribution: Knowledge of the physical connections and/or cause effect relationships between the ac distribution system and the following systems: DC distribution	**Tier**	**2**		
	Group	**1**		
	K/A	**062 K1.03**		
	IR	**3.5**		

Question 42

Regarding the interface between Class AC and DC power sources…

(1) Each Class Swing Battery Charger has a total of _____ 480V Motor Control Center(s) available to supply power to the AC Input of the charger

(2) Each Class 125 VDC Bus has a total of _____ Battery Chargers which are available to be aligned to the bus

A. (1) 1
(2) 2

B. (1) 1
(2) 3

C. (1) 2
(2) 2

D. (1) 2
(2) 3

Proposed Answer:	A

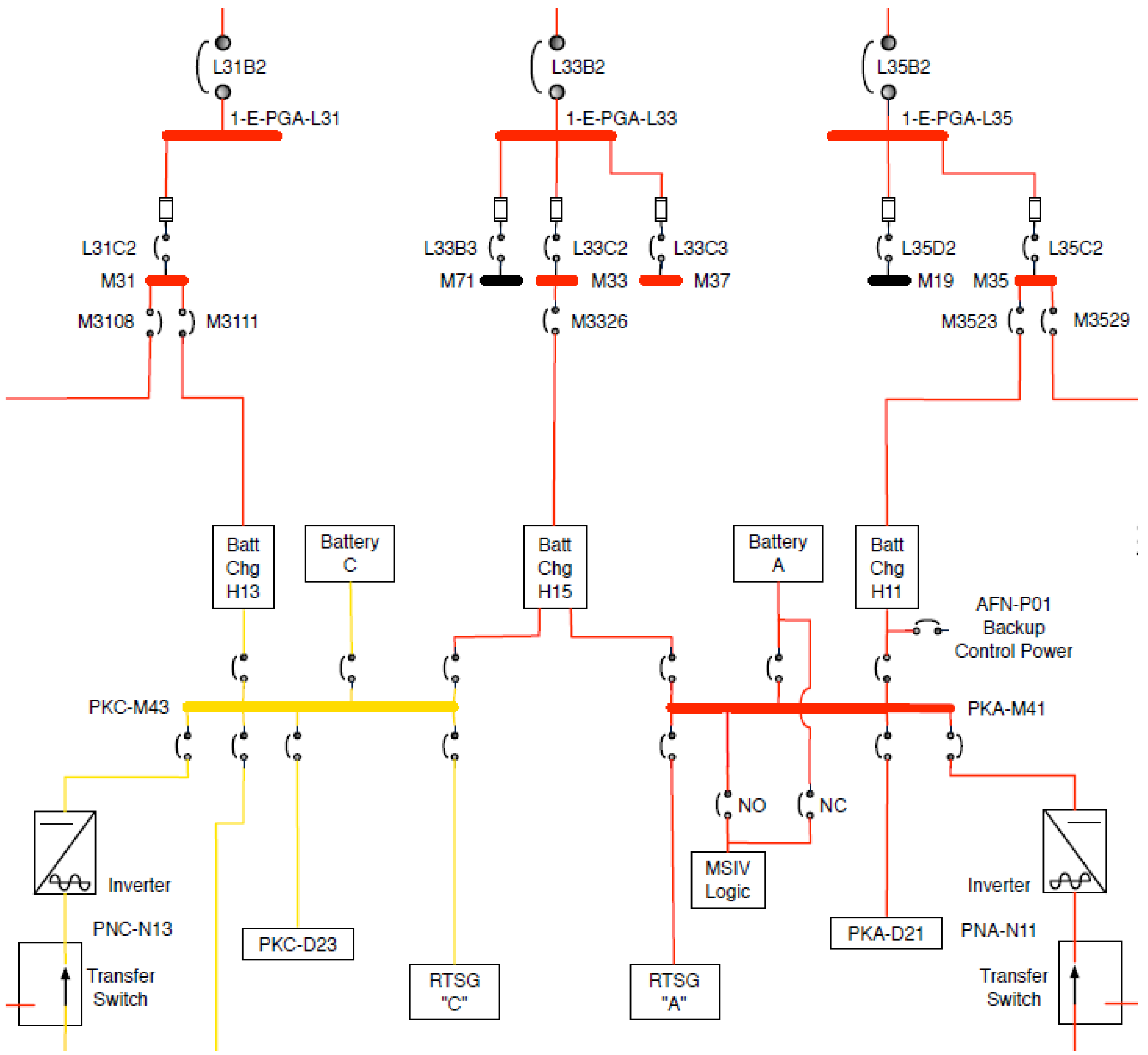
Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible as Non-Class 125 DC Bus, NKN-M45 has three chargers which can each be aligned to the bus, however the class DC buses each only have two.
C.	First part is plausible since each swing charger can output to one of two DC buses, however they can only receive AC power from one loadcenter. Additionally, each Class 480 MCC is powered from a Class 4kV Bus, which each have multiple sources of power, but the Battery Charger only has one available source of AC power. Second part is correct.
D.	First part is plausible since each swing charger can output to one of two DC buses, however they can only receive AC power from one loadcenter. Additionally, each Class 480 MCC is powered from a Class 4kV Bus, which each have multiple sources of power, but the Battery Charger only has one available source of AC power. Second part is plausible as Non-Class 125 DC Bus, NKN-M45 has three chargers which can each be aligned to the bus, however the class DC buses each only have two.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

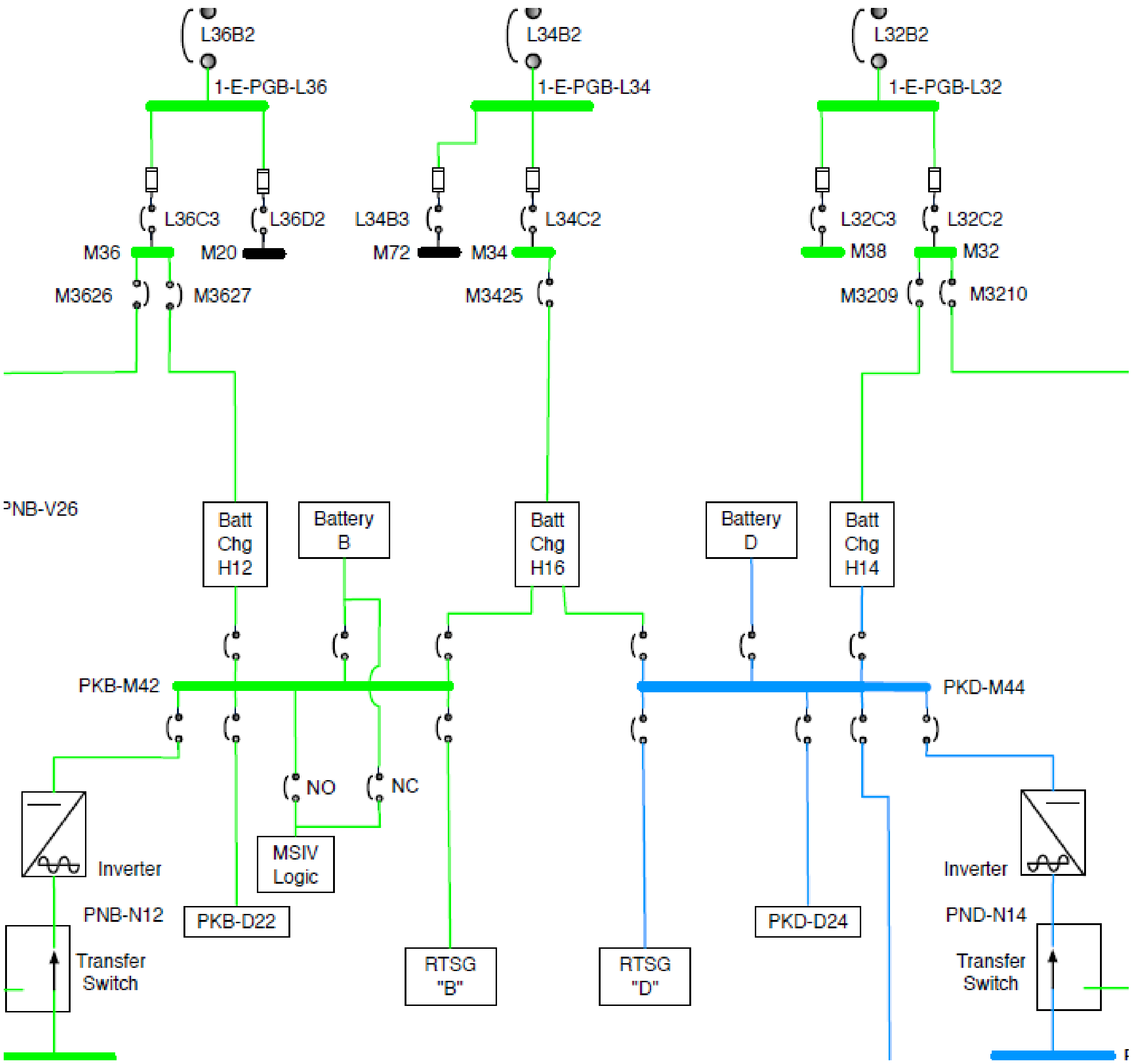
Level of Difficultly:	3	
10CFR55.41:	7	
Reference Provided:	N	
Learning Objective:	Draw a simplified diagram of the Class 120 VAC Electrical Distribution System with Ametek inverters	

Train A DC Power – Charger H15 is the swing charger:



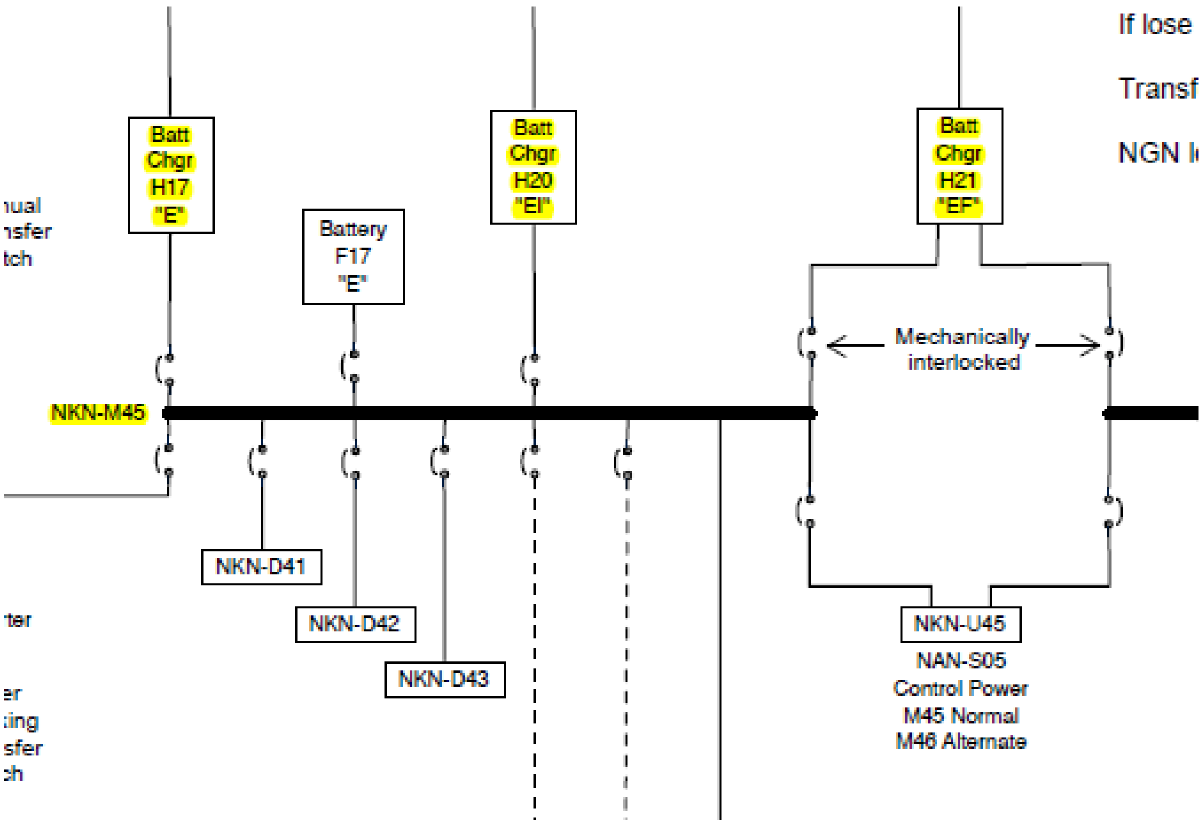
Technical Reference:	PVNGS Operator Information Manual

Train B DC Power, Charger H16 is the swing charger:



Technical Reference:	PVNGS Operator Information Manual

Non-Class DC Bus NKN-M45 has 3 chargers which can be aligned to it:



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: DC Electrical Distribution: Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Grounds	**Tier**	**2**		
	Group	**1**		
	K/A	**063 A2.01**		
	IR	**2.5**		

Question 43

Given the following conditions:

- Unit 3 is operating at 100% power

Subsequently:

- The Reactor was tripped due to a complete loss of condenser vacuum
- On the Trip, a ground on PKA-M41 resulted in PKA-M41 being de-energized

Prior to ANY operator action being taken in the Control Room or in the field, which of the following describes the impact of the loss of PKA-M41 on the availability of AFN-P01?

The suction valves for AFN-P01, CTA-HV-1 and CTA-HV-4, ___(1)___ and the feeder breaker for AFN-P01 ___(2)___ .

A. (1) can be opened from the Control Room
(2) can be closed from the Control Room

B. (1) can be opened from the Control Room
(2) must be closed locally at the breaker

C. (1) must be opened locally at the valves
(2) can be closed from the Control Room

D. (1) must be opened locally at the valves
(2) must be closed locally at the breaker

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible since AFN-P01 has an alternate control power source which will allow for breaker operation from the control room with PKA de-energized, however the control power shift must be taken in the field
B.	Correct.
C.	First part is plausible since this would be true if the suction valves for AFN were powered from PK, however they are powered from PH. Second part is plausible since AFN-P01 has an alternate control power source which will allow for breaker operation from the control room with PKA de-energized, however the control power shift must be taken in the field
D.	First part is plausible since this would be true if the suction valves for AFN were powered from PK, however they are powered from PH. Second part is correct.

Question Source:	x	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		**Memory or Fundamental Knowledge**
	x	**Comprehension or Analysis**

Level of Difficultly:	3	
10CFR55.41:	4	
Reference Provided:	N	
Learning Objective:	**Describe the Control Room controls associated with the Non Essential Auxiliary Feedwater Pump AFN-P01 including its indications.**	

Technical Reference:	40OP-9PH01, Train A 480V Class 1E MCC

Appendix C - PHA-M33, 480V Class 1E MCC

Number	Name	Location	Drawing	Required Position	Positioned By Initials & Date	Verified By Initials & Date
PHA-M3302	Ckt Brk for Spr Pond Pump Hse Exhst Fan Motor	Aux Bldg, 120', West Elect Pen Rm	E-PHA-003	On		
PHA-M3303	FLEX Feed to RCS Makeup Pump	Aux Bldg, 120', West Elect Pen Rm	E-PHA-003	Locked Off per 40AC-0ZZ06		
PHA-M3304	Ckt Brk for HPSI Flow Control Valve SIA-UV-647	Aux Bldg, 120', West Elect Pen Rm	E-PHA-003	On		
PHA-M3305	SIA-HV-604 Ckt Brk "A" HPSI Pmp Long Trm Clg Vlv	Aux Bldg, 120', West Elect Pen Rm	E-PHA-003	On		
PHA-M3306	Radiation Ckt Brk for SQN-D01 Dist Panel	Aux Bldg, 120', West Elect Pen Rm	E-PHA-003	On		
PHA-M3306 50G	Relay Ground Fault for Distribution Panel	Aux Bldg, 120', West Elect Pen Rm	E-PHA-003	Reset		
PHA-M3307	Ckt Brk Backup for E-PHA-M3306	Aux Bldg, 120', West Elect Pen Rm	E-PHA-003	On		
PHA-M3307 50G	Relay Ground Fault Backup for E-PHA-M3306	Aux Bldg, 120', West Elect Pen Rm	E-PHA-003	Reset		
PHA-M3308	CTA-P01 "A" Cond Xfer Pump	Aux Bldg, 120', West Elect Pen Rm	E-PHA-003	On		
PHA-M3309	Ckt Brk for AFN-P01 Suction Isol Valve CTA-HV-1	Aux Bldg, 120', West Elect Pen Rm	E-PHA-003	On		

Technical Reference:	40OP-9PH01, Train A 480V Class 1E MCC

Appendix E - PHA-M35, 480V Class 1E MCC

Number	Name	Location	Drawing	Required Position	Positioned By Initials & Date	Verified By Initials & Date
PHA-M3503	Ckt Brk for Shutdn Clg Iso Loop 1 Vlv SIA-UV-651	Aux Bldg, 120', West Elect Pen Rm	E-PHA-005	On		
PHA-M3504	SIA-UV-655 S/D Clng Cntmt Isol Vlv	Aux Bldg, 120', West Elect Pen Rm	E-PHA-005	On		
PHA-M3505	Ckt Brk for AFN-P01 Suction Isol Valve CTA-HV-4	Aux Bldg, 120', West Elect Pen Rm	E-PHA-005	On		
PHA-M3506	SIA-UV-664 Ckt Brk "A" CS Pump Recirc Vlv	Aux Bldg, 120', West Elect Pen Rm	E-PHA-005	On		

Technical Reference:	LOIT Aux Feedwater Lesson Plan

Main Idea

Loss of Class 125VDC bus PKA-M41/PKA-D21.

AFA-P01

- Results in a loss of governor control power.
- The turbine trips on overspeed if running.
- Steam Supplies Fail as is.
- AF isolation valves Fail as is.

AFN-P01 Loss of DC control power

- Alternate control power supply from battery charger "A" via 3 position switch E-PBA-U01 on Train "A" 4KV switchgear

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Emergency Diesel Generator: Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Fuel oil storage tanks	**Tier**	**2**		
	Group	**1**		
	K/A	**064 K6.08**		
	IR	**3.2**		

Question 44

Given the following conditions:

- The ‘B’ EDG is operating at rated load
- An auto makeup to the ‘B’ Fuel Oil Day Tank is in progress

Subsequently:

- The associated Fuel Oil Transfer Pump tripped
- The ‘B’ EDG Fuel Oil Day Tank level is currently 650 gallons

Approximately how much longer can the ‘B’ EDG continue to operate at rated load?

A. ~ 25 minutes

B. ~ 50 minutes

C. ~ 100 minutes

D. ~ 200 minutes

Proposed Answer:	C

Explanations:	
A.	Plausible if thought that the EDG consumes 25 gpm at rated load (650 gallons / 25 gpm = 26 minutes), however 25 gpm is the capacity of the Fuel Oil Transfer Pump, not fuel consumption at rated load
B.	Plausible if thought that 650 gallons in the day tank could provide 100 minutes of total run time for both EDGs, thus could only provide 50 minutes of operation for each EDG
C.	Correct. At rated load, the EDG consumes ~ 6.5 gpm, therefore 650 gallons = 100 minutes
D.	Plausible if thought that 650 gallons could provide for each EDG to run for 100 minutes, thus could provide a single EDG at rated load for 200 minutes.

Question Source:		New	
		Bank	
	x	**Modified**	
	x	**Previous NRC Exam**	**2019 Q41 (modified tank level to change answer)**

Cognitive Level:		Memory or Fundamental Knowledge
	x	**Comprehension or Analysis**

Level of Difficultly:	3	
10CFR55.41:	**7**	
Reference Provided:	**N**	
Learning Objective:	**Discuss the purpose and conditions under which the Diesel Generator System is designed to function.**	

Technical Reference:	LOIT Emergency Diesel Generators Lesson Plan

650 gallons / 6.5 gpm = ~ 100 minutes

Fuel Oil Day Tanks

- Each DG has its own Day Tank.
- Each day tank has a usable capacity of about 1055 gallons.
- Minimum level is 2.75' (~550 gals) by Technical Specification 3.8.1, which exceeds the requirement of a sixty minute fuel supply with 10% margin assuming full rated load on the DG, which is 438 gallons.
- The transfer pump will automatically start to refill the day tank when level decreases to approximately 3.2 ft. (63%). At that time there are 728 gals of fuel oil, which can sustain the DG full load operation for about 1.9 hours. The pump stops at approximately 4.6 ft (93%).
- The DG uses approximately 6.55 GPM at full rated load.
- Each tank is equipped with an overflow and drain connection routed to the storage tank.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Process Radiation Monitoring: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	**Tier**	**2**		
	Group	**1**		
	K/A	**073 G 2.4.50**		
	IR	**4.1**		

Question 45

Given the following conditions:

- Unit 2 is operating at 100% power
- A SGTL is in progress on SG #1
- RU-139, Main Steam Line – SG #1, is in HIGH ALARM
- RU-141, Condenser Vacuum / Gland Seal Exhaust, is in HIGH ALARM
- Both alarms have been confirmed to be valid

Per 74AL-9SQ01, Radiation Monitoring System Alarm Validation and Response, the crew should ___(1)___ in response to the high alarm on RU-139, and should ___(2)___ in response to the high alarm on RU-141.

A. (1) secure blowdown from SG #1
   (2) perform 40DP-9ZZ14, Contaminated Water Management

B. (1) secure blowdown from SG #1
   (2) ensure the Post Filter Mode Select Switch is in the THRU FILTER MODE

C. (1) ensure AFA-P01 is not running
   (2) perform 40DP-9ZZ14, Contaminated Water Management

D. (1) ensure AFA-P01 is not running
   (2) ensure the Post Filter Mode Select Switch is in the THRU FILTER MODE

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible since this action is directed in the Excessive RCS Leakreate (for SGTL) AOP and is a logical action to take in response to high activity in the Main Steam line, however this is not directed in the RM ARP.
B.	Correct.
C.	First part is plausible since use of AFA-P01 with a SGTL in progress creates a direct release to the environment, however this action is not directed in the RM ARP. Second part is plausible since this action is directed in the Excessive RCS Leakreate (for SGTL) AOP and is a logical action to take in response to high activity in the Main Steam line, however this is not directed in the RM ARP.
D.	First part is plausible since use of AFA-P01 with a SGTL in progress creates a direct release to the environment, however this action is not directed in the RM ARP. Second part is correct.

Question Source:		New	
	x	Bank	
		Modified	
	x	Previous NRC Exam	2016 NRC Q11

Cognitive Level:	x	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	11	
Reference Provided:	N	
Learning Objective:	Describe Operation's responsibilities for RMS alarms	

Technical Reference:	74AL-9SQ01, Radiation Monitoring System Alarm Validation and Response

___ 1. Verify the alarm is valid.

___ 2. IF the alarm is valid,
THEN perform the following:

___ 2.1 Perform actions per 40AO-9ZZ02, Excessive RCS Leakrate.

___ 2.2 IF RU-139 is currently in alarm,
THEN secure blowdown to Steam Generator #1 per 40OP-9SG03, Operating the Steam Generator Blowdown System.

Technical Reference:	74AL-9SQ01, Radiation Monitoring System Alarm Validation and Response

___ 1. IF ANY of the following:

- A HIGH alarm is received on Channel 1 of RU-141
- A HIGH alarm is received on Channel 2 of RU-141

THEN ensure ARN-HS-19, Post Filter Mode Selector Switch, is in the "THRU FILTER MODE."

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Service Water: Knowledge of the effect that a loss or malfunction of the SWS will have on the following: ESF loads	**Tier**	**2**		
	Group	**1**		
	K/A	**076 K3.07**		
	IR	**3.7**		

Question 46

Per 40OP-9DG01, Emergency Diesel Generator A, the MAXIMUM safe operating time for the 'A' EDG following a loss of Spray Pond water is ___(1)___ if operating at FULL load, and ___(2)___ if operating at NO load.

A. (1) 1.5 minutes
   (2) 15 minutes

B. (1) 1.5 minutes
   (2) 30 minutes

C. (1) 2.6 minutes
   (2) 15 minutes

D. (1) 2.6 minutes
   (2) 30 minutes

Proposed Answer:	C

Explanations:	
A.	First part is plausible if thought that the time limit at full load would be 10% of the time limit at no load, however the actual full load time limit is 2.6 minutes. Second part is correct.
B.	First part is plausible if thought that the time limit at full load would be 5% of the time limit at no load (if assuming the no load limit was 30 minutes), however the full load time limit is 2.6 minutes. Second part is plausible as the time limit until damage occurs for an RCP with no cooling flow is 30 minutes, however for the EDG it is 15 minutes.
C.	Correct.
D.	First part is correct. Second part is plausible as the time limit until damage occurs for an RCP with no cooling flow is 30 minutes, however for the EDG it is 15 minutes.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	7	
Reference Provided:	N	
Learning Objective:	Discuss the purpose and conditions under which the Essential Spray Pond System 'SP' is designed to function.	

Technical Reference:	40OP-9DG01, Emergency Diesel Generator A

PALO VERDE PROCEDURE		Page 8 of 185
Emergency Diesel Generator A	40OP-9DG01	Revision 80

3.0 PRECAUTIONS AND LIMITATIONS

3.1 Precautions

3.1.1 Transferring the voltage regulator control mode while the Diesel Generator is operating will cause a current surge resulting in equipment damage.

3.1.2 If either control power supply (DC Power On circuit 1 & 2 control power lights on DGA-B01) to the DG is lost, the DG output breaker may NOT trip if the Diesel Generator should trip during parallel operation. Under these circumstances, PBA-S03B, A DG Output Breaker, must be tripped manually to prevent motorizing Train A Diesel Generator.

3.2 Limitations

3.2.1 DGA-HS-7, DG "A" Mode Control, shall NOT be placed to OFF except for maintenance. Placing DGA-HS-7, DG "A" Mode Control, to OFF will prevent Train A Diesel Generator from starting in either automatic or manual modes.

3.2.2 When Bus PBA-S03 is being supplied from the alternate ESF Transformer (NBN-X04), electrical interlocks prevent paralleling with the Emergency Diesel Generator PEA-G01.

3.2.3 NO smoking or open flames are permitted in the vicinity of the Fuel Oil System.

3.2.4 Safe operating time periods following a loss of Spray Pond water:

- Full load - 2.6 minutes
- Zero load - 15 minutes

Technical Reference:	LOIT Loss of Cooling Water Lesson Plan

Main Idea

Upon a loss of NC (cooling water) to the RCP(s), operators have thirty (30) minutes to reduce power or isolate cooling water and shutdown the RCP(s). If an RCP is allowed to operate more than 30 minutes without cooling water, possible pump motor assembly bearing seizure may occur.

Explanation

This objective is linked to other lessons.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Instrument Air: Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: MSIV air	Tier	2		
	Group	1		
	K/A	078 K1.05		
	IR	3.4		

Question 47

Given the following conditions:

- Unit 2 is operating at 100% power
- An Instrument Air rupture has occurred just downstream of the IA compressors
- IA pressure is at atmospheric pressure throughout the system
- The nitrogen backup supply valve has failed closed

Based on these conditions, the Main Steam Isolation Valves will…

A. slow close due to the loss of IA

B. fast close due to the loss of IA

C. remain open and can ONLY be slow closed

D. remain open and can ONLY be fast closed

Proposed Answer:	D

Explanations:	
A.	Plausible that the MSIVs would fail closed as this is the fail safe position, and the valves are stroked open in slow speed and can be closed in slow speed, however the MSIVs remain open on a loss of instrument air
B.	Plausible that the MSIVs would fail closed as this is the fail safe position, and the valves are normally closed in fast speed, however the MSIVs remain open on a loss of instrument air.
C.	Plausible since the MSIVs will remain open, however slow close is not available on a loss of instrument air.
D.	Correct.

Question Source:		New	
	x	Bank	
		Modified	
	X	Previous NRC Exam	2019 NRC Q44

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	4	
Reference Provided:	N	
Learning Objective:	Determine the major effects on plant operation as instrument air pressure degrades.	

Appendix A, Expected Component Failure as System Pressure Drops

PRESS	COMPONENT	ACTION
60 - 50 psig ED	EDN-BTV-3 / 4 / 13 / 14 / 23 / 24 / 59 / 60 / 61 / 69 / 70 / 71 / 73 / 74 / 75, Bleeder Trip Valves (FC)	**NOTE** ARDV-1, Main Turbine Front Standard Turbine Trip Air Relay Dump Valve actuates EDNPSL76 closing the Bleeder Trip valves. Extraction steam flow will maintain the valves in the open position until pressure decays, allowing the valves to close preventing any backflow from the heaters to the turbine.
NC	NCN-LV-75, Nuclear Cooling Water Surge Tank Demin Water Makeup Valve (FC)	____ 1. **IF** makeup will be provided to the NC Surge Tank, **THEN** PERFORM 40OP-9NC01, Nuclear Cooling Water (NC), Alternate Makeup to NC System, to maintain normal level in the NC Surge Tank.
SG	SGE-UV-170 / 171 / 180 / 181, MSIV (FAIL AS IS)	**NOTE** Fast closure operation is available via the accumulator, slow mode valve operation will not be available.
		____ 1. **IF** the MSIVs will be closed, **THEN** fast close the MSIVs using **ANY** of the following: **SG #1** • SGA-HS-251 • SGB-HS-253 **SG #2** • SGA-HS-250 • SGB-HS-252

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Containment: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system controls including: Containment pressure, temperature, and humidity	**Tier**	**2**		
	Group	**1**		
	K/A	**103 A1.01**		
	IR	**3.7**		

Question 48

The Containment Spray System is designed such that a single train of Containment Spray will reduce the peak Containment pressure following a design basis accident by a MINIMUM of ___(1)___ % of peak Containment pressure within a MAXIMUM of ___(2)___ hours.

A. (1) 25
(2) 24

B. (1) 25
(2) 48

C. (1) 50
(2) 24

D. (1) 50
(2) 48

Proposed Answer:	C

Explanations:	
A.	First part is plausible if thought that two trains will reduce pressure 50% in 24 hours, however each train is independently able to reduce pressure 50% in 24 hours. Second part is correct.
B.	First part is plausible if thought that two trains will reduce pressure 50% in 24 hours, however each train is independently able to reduce pressure 50% in 24 hours. Second part is plausible if thought that with only one train available pressure would be reduced by half and it would take double the time.
C.	Correct.
D.	First part is correct. Second part is plausible if thought that with only one train available, 50% reduction would take double the time.

Question Source:	x	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	x	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	8	
Reference Provided:	N	
Learning Objective:	Describe the design basis associated with the Containment Spray system.	

Technical Reference:	LOIT Containment HVAC Lesson Plan

Main Idea

The Containment Spray System is designed to provide for the following:

- Prevent exceeding containment design pressure and temperature limits (60 psig and 300°F).
- Mitigate the consequences of any size break. (See introduction)
- Reduce containment pressure and temperature and maintain them at acceptable levels during recirculation operations.
- For the containment design basis accident, the containment spray system is designed to reduce containment pressure from peak value to one-half peak value in less than 24 hours.
- Consists of two redundant and independent trains each of which provides 100% of the required heat removal capability and 100% of the required iodine removal capability.
- The portions of the system located inside containment are designed to remain operable in the post accident environment.
- Deliver flow to the Shutdown Cooling System at a head which is compatible with the Shutdown Cooling System to augment LPSI pump flow.

  When the RCS temperature is below 200ºF and pressurizer pressure less than 250 psia spray pumps may be realigned and started to provide additional flow through SDC heat exchangers

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Reactor Coolant Pump: Ability to manually operate and/or monitor in the control room: RCP motor parameters	Tier	2	[redacted]	
	Group	1	[redacted]	
	K/A	003 A4.02		
	IR	2.9	[redacted]	

Question 49

Given the following conditions:

- Unit 2 is operating at 100% power
- Seal Injection Containment Isolation Valve, CHN-HV-255, has just failed closed and cannot be reopened from Board 3

Assuming no operator action, what will be the effect on the Reactor Coolant Pump System?

RCP HP Seal Cooler Inlet temperature will ___(1)___ and all other seal temperatures monitored on Board 4, will ___(2)___ .

A. (1) exceed 250°F
(2) remain normal

B. (1) exceed 250°F
(2) exceed 200°F

C. (1) stabilize between 200 and 220°F
(2) remain normal

D. (1) stabilize between 200 and 220°F
(2) exceed 200°F

Proposed Answer:	C

Explanations:	
A.	Plausible that trip criteria of 250°F would be exceeded since HPSC inlet temp is the outlet temp from the RCP journal bearing and the loss of seal injection results in a partial loss of cooling to the seals, however with NC still in service, HPSC inlet temp will stabilize between 200 and 220°F. Second part is correct.
B.	Plausible that trip criteria of 250°F would be exceeded since HPSC inlet temp is the outlet temp from the RCP journal bearing and the loss of seal injection results in a partial loss of cooling to the seals, however with NC still in service, HPSC inlet temp will stabilize between 200 and 220°F and all other seal temps will rise but remain in their normal control bands.
C.	Correct.
D.	First part is correct. Plausible that trip criteria of 200°F would be exceeded since the loss of seal injection results in a partial loss of cooling to the seals, however with NC still in service all other seal temps will rise but remain in their normal control bands.

Question Source:		New	
	X	Bank	
		Modified	
	x	Previous NRC Exam	2016 NRC Q28

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	3	
Reference Provided:	N	
Learning Objective:	Explain the operation of the RCP Seal Injection Header Isolation Valve (CHB-HV-255), including the Control Room control, under normal operating conditions.	

Technical Reference:	40AO-9ZZ04, RCP Emergencies

4.0 ABNORMAL RCP SEAL PARAMETERS

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 1. Enter AOP Entry Time and Date: ______________________	

-- **NOTE** --

RCP HP Seal Cooler inlet temperature should rise to between 200°F and 220°F if seal injection is stopped. All other seal temperatures should remain normal.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Residual Heat Removal: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: Closed cooling water flow rate and temperature	Tier	2		
	Group	1		
	K/A	005 A1.03		
	IR	2.5		

Question 50

Given the following conditions:

- Unit 1 is in MODE 4
- The crew is placing SDC in service using the Train ‘A’ LPSI Pump
- The CRS directs the OATC to ensure they are warming up the Train ‘A’ SDCHX at the MAXIMUM allowable heat up rate per 40OP-9SI01, Shutdown Cooling Initiation
- Currently:
  - The ‘A’ LPSI Pump is running
  - SIA-UV-635, LPSI Header A to RC Loop 1A, is 10% open
  - SIA-HV-306, LPSI S/D Cooling HX A Bypass Valve, is 20% open

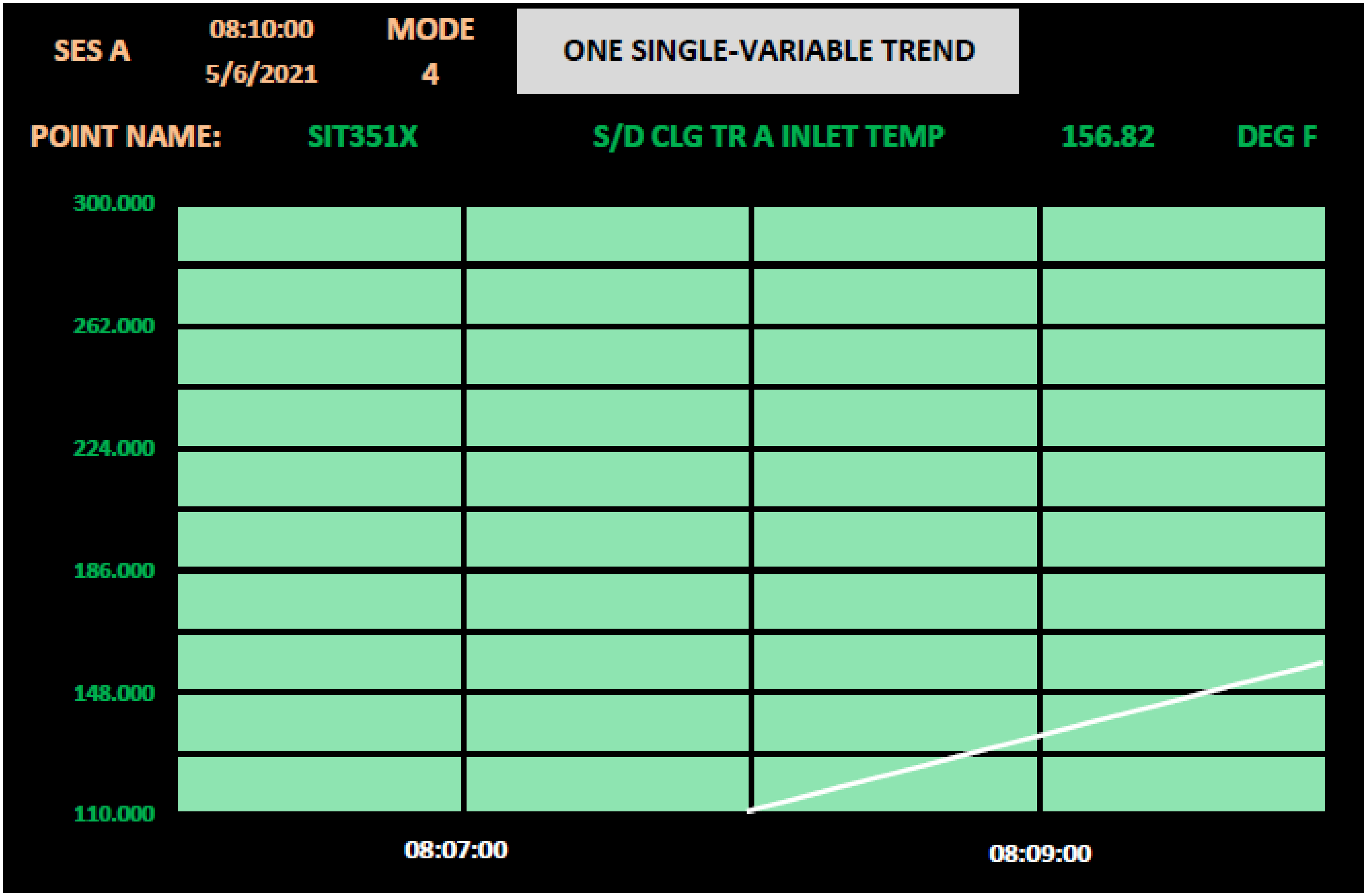
Based on the provided trend on the following page, in order to comply with the CRS direction, the ‘A’ SDCHX heat up rate should be ___(1)___ and the crew can accomplish this by throttling ___(2)___ on SIA-HV-306, LPSI S/D Cooling HX A Bypass Valve.

A. (1) raised
   (2) open

B. (1) raised
   (2) closed

C. (1) lowered
   (2) open

D. (1) lowered
   (2) closed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S/D CLG TR A INLET TEMP
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DEG F
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262.000
224.000
186.000
148.000
110.000
08:07:00
08:09:00

Proposed Answer:	C

Explanations:	
A.	First part is plausible if either the maximum heat up rate is unknown to the examinee (19°/min) or if the either the X or Y-axis is misinterpolated, however the graph shows a current heat up rate of 23°F/min so the rate needs to be lowered. Second part is correct.
B.	First part is plausible if either the maximum heat up rate is unknown to the examinee (19°/min) or if the either the X or Y-axis is misinterpolated, however the graph shows a current heat up rate of 23°F/min so the rate needs to be lowered. Second part is plausible since closing 306 would raise the heatup rate, however in this case the heatup rate needs to be lowered.
C.	Correct.
D.	First part is correct. Second part is plausible since HV-306 is throttled closed to raise the cooldown rate of the RCS, however to lower the heatup rate of the SDCHX, HV-306 must be throttled open.

Question Source:		New	
		Bank	
	X	Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	3	
Reference Provided:	Y	Attached picture of the SDC Train ‘A’ Inlet Temperature
Learning Objective:	Describe the Control Room indications associated with the SDC system.	

Original Question:	Original question had picture with heatup rate of 13°F/min so the answer was different.

Given the following conditions:

- Unit 1 is in MODE 4
- The crew is placing SDC in service using the Train ‘A’ LPSI Pump
- The CRS directs warming up the Train ‘A’ SDCHX at the MAXIMUM heat up rate allowed by 40OP-9SI01, Shutdown Cooling Initiation
- SIA-HV-306, LPSI S/D Cooling HX A Bypass Valve, is 20% open
- The ‘A’ LPSI Pump has been started
- SIA-UV-635, LPSI Header A to RC Loop 1A, is 10% open

Based on the trend on the following page, in order to comply with the CRS direction, the ‘A’ SDCHX heat up rate should be ___(1)___ and the crew can accomplish this by throttling ___(2)___ on SIA-HV-306, LPSI S/D Cooling HX A Bypass Valve.

A. (1) raised
(2) open

B. (1) raised
(2) closed

C. (1) lowered
(2) open

D. (1) lowered
(2) closed

Technical Reference:	40OP-9SI01 Shutdown Cooling Initiation

Step 6.15.23, Continued

___ I. Throttle SIA-HV-306 to achieve ALL of the following using handswitch SIA-HS-306, LPSI S/D Cooling HX A Bypass Vlv HV-306:

- ___ The flow rate determined in Step 6.15.23.G
- ___ RCS cooldown rate determined in Step 6.15.23.D.1
- ___ SIA-E01, Shutdown Cooling Heat Exchanger 1, heatup rate less than 19°F/ minute

3.0 PRECAUTIONS AND LIMITATIONS

3.1 Precautions

3.1.1 Exceeding a 19°F per minute heatup or cooldown rate on the SDC loop may cause damage to the SDC heat exchanger.

Technical Reference:	SI System Training Manual

Throttling OPEN SIA-HV-306 (SDCHX A Bypass valve) will reduce the amount of flow through the HX. This will result in a lower heatup rate on the HX.

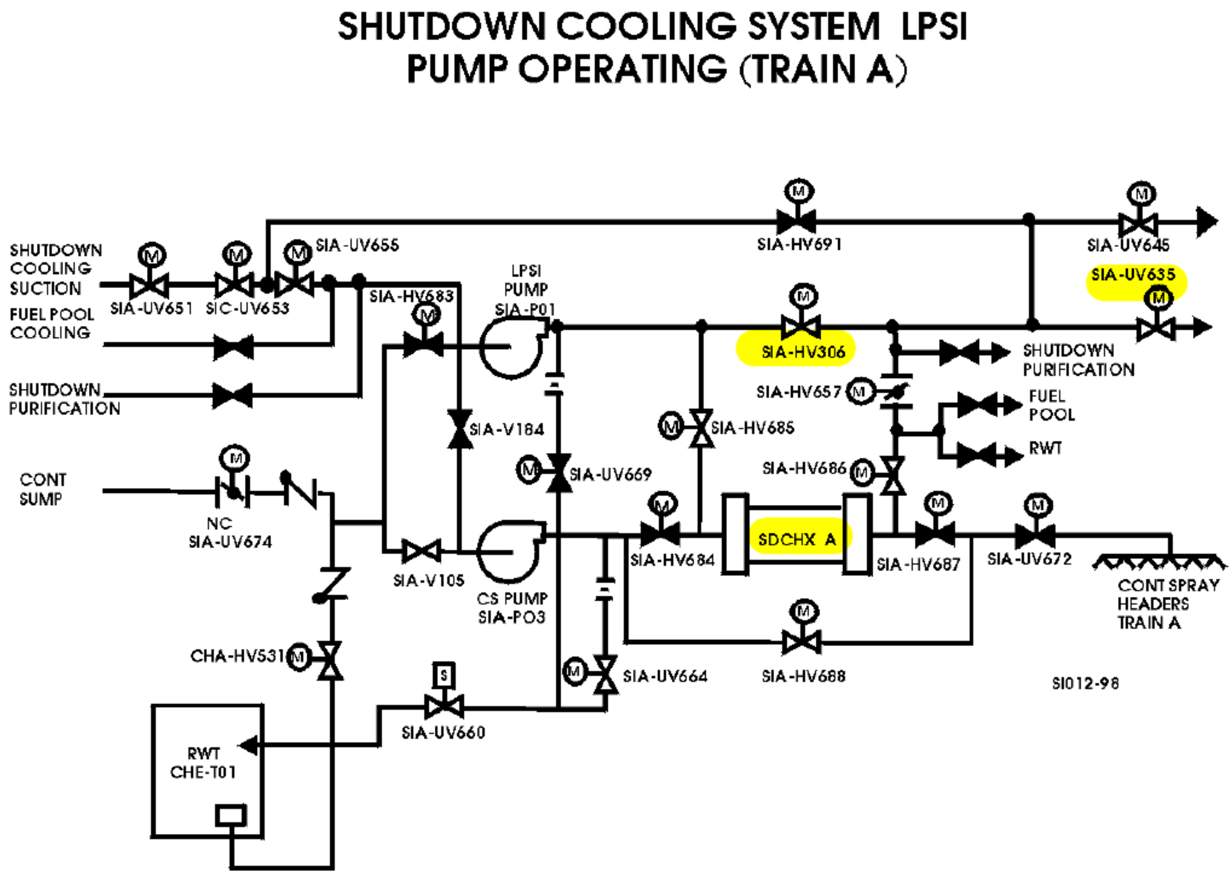


Figure 3 - 1 Shutdown Cooling System - LPSI Pump A Operating

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Pressurizer Pressure Control: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PZR PCS controls including: RCS pressure	Tier	2		
	Group	1		
	K/A	010 A1.07		
	IR	3.7		

Question 51

Given the following conditions:

- Unit 3 is operating at 100% power
- The Pressurizer is in Boron Equalization with all Non-Class Backup Heaters in service
- RCN-PIC-100, Pressure Master Control, is in AUTO with a setpoint of 2220 psia

Subsequently:

- A transient occurred causing RCS pressure to RISE

Assuming the PPCS system is functioning normally and that RCS pressure continues to rise, the Main Spray Valves will be FULL OPEN…

___(1)___ the Backup Heaters receive a trip signal,

AND

___(2)___ the high Pressurizer pressure TS limit is exceeded.

A. (1) BEFORE
   (2) BEFORE

B. (1) BEFORE
   (2) AFTER

C. (1) AFTER
   (2) BEFORE

D. (1) AFTER
   (2) AFTER

Proposed Answer:	A

Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible since during normal operations (non-boron equalization) the spray valves are full open at 2300 psia, however the pressure at which the spray valves are full open is dependent on the setpoint of PIC-100, so in this condition, the spray valves will be full open prior to exceeding the TS limit of 2295 psia.
C.	First part is plausible since during normal operations, the heaters trip at 2285 psia and the main spray valves are full open at 2300 psia, however the heater trip is a firm setpoint and not dependent on the setpoint of PIC-100 so with the setpoint at 2220 psia, the spray valves will be full open prior to the heaters receiving a trip signal. Second part is correct.
D.	First part is plausible since during normal operations, the heaters trip at 2285 psia and the main spray valves are full open at 2300 psia, however the heater trip is a firm setpoint and not dependent on the setpoint of PIC-100 so with the setpoint at 2220 psia, the spray valves will be full open prior to the heaters receiving a trip signal. Second part is plausible since during normal operations (non-boron equalization) the spray valves are full open at 2300 psia, however the pressure at which the spray valves are full open is dependent on the setpoint of PIC-100, so in this condition, the spray valves will be full open prior to exceeding the TS limit of 2295 psia.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	5	
Reference Provided:	N	
Learning Objective:	Describe the automatic features associated with the Pressurizer Pressure Control System Bistables.	

Technical Reference:	40OP-9CH01 CVCS Normal Operations

PZR master pressure controller is set to 30 psi below NOP (~2220 psia) during boron equilization.

Appendix N - Equalizing Pressurizer Boron Concentration

1.0 STARTING PRESSURIZER BORON EQUALIZATION

___ 1.5 Lower RCN-PIC-100, Pressure Master Control, setpoint 30 psi less than current RCS pressure.

Technical Reference:	Technical Specifications

Upper pressure limit for LCO 3.4.1 is 2295 psia.

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, cold leg temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ≥ 2130 psia and ≤ 2295 psia; and

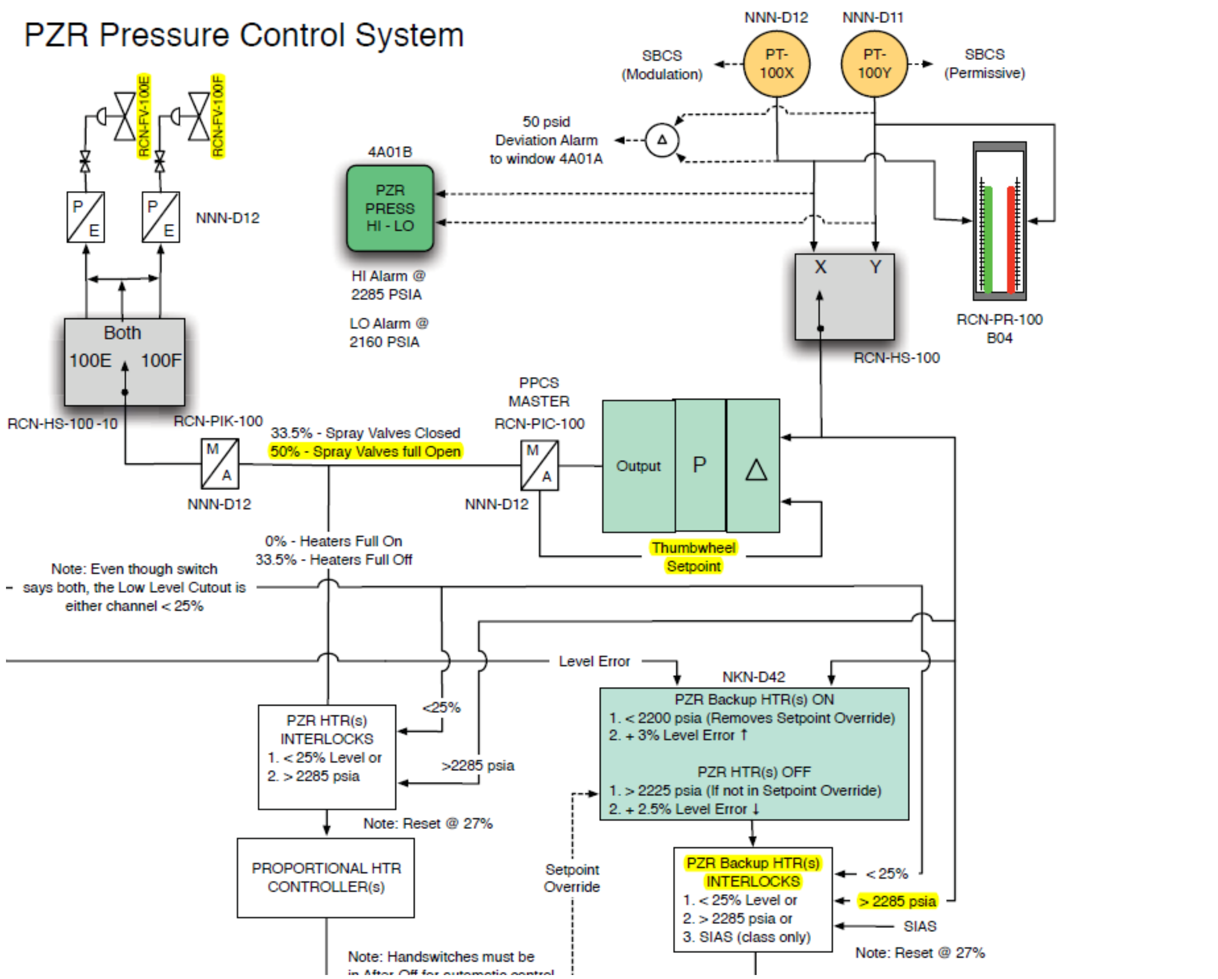
b. RCS cold leg temperature (T_c) shall be within the area of acceptable operation shown in Figure 3.4.1-1; and

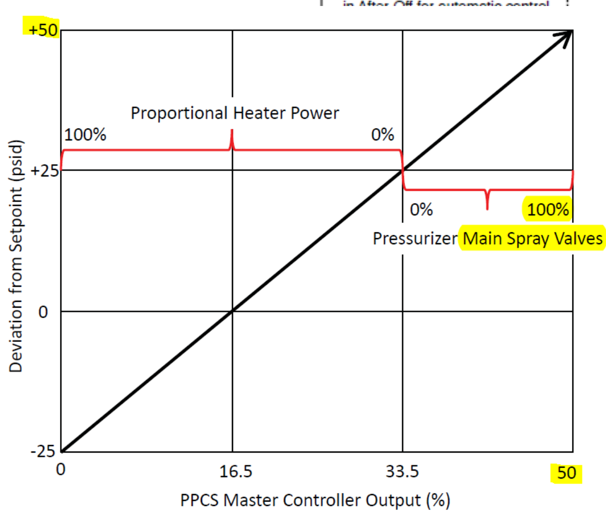
c. RCS total flow rate ≥ 155.8 E6 lbm/hour.

APPLICABILITY: MODE 1 for RCS total flow rate,

MODES 1 and 2 for pressurizer pressure,

Spray valves operate based on the output of PPCS Master controller. With the thumbwheel setpoint set to 2220 psia, spray valves will be full open at 2270 psia. The high pressure heater cutout (2285 psia) setpoint does not change. It is operated from a bistable based on the output of the selected PZR pressure transmitter.





Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Reactor Protection: Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of dc control power	Tier	2		
	Group	1		
	K/A	012 A2.07		
	IR	3.2		

Question 52

Given the following conditions:

- Unit 2 is operating at 100% power
- A loss of PKC-M43 has just occurred

Per 40AO-9ZZ13, Loss of Class Instrument or Control Power, the crew should…

A. manually open ONLY the ‘C’ RTCB

B. manually open the ‘A’ AND the ‘C’ RTCBs

C. verify that ONLY the ‘C’ RTCB automatically opened

D. verify that the ‘A’ AND the ‘C’ RTCBs automatically opened

Proposed Answer:	C

Explanations:	
A.	Plausible since only the 'C' RTCB should be open, and on a loss of control power breakers are normally required to be manually operated, however on a loss of PKC-M43, the 'C' RTCB will automatically open.
B.	Plausible since on a loss of PNC-D27 both the 'A' and 'C' RTCBs will open, and normally a loss of control power requires local manual breaker operation, however on a loss of DC control power, only the 'C' breaker will open.
C.	Correct.
D.	Plausible since this is the case on a loss of PNC-D27, however on a loss of PKC-M43, only the 'C' RTCB will open.

Question Source:	x	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	x	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	Given a loss of PK and/or PN, describe how the RPS responds to the power loss in accordance with 40AO-9ZZ13.	

Technical Reference:	40AO-9ZZ13 Loss of Class Instrument or Control Power

The AOP will direct the crew to review appendix E for impacts of the loss of PKC-M43. The crew will then verify that the plant responded as expected (i.e. verify C RTSG breaker open).

7.0 LOSS OF PKC-M43 OR PKC-D23

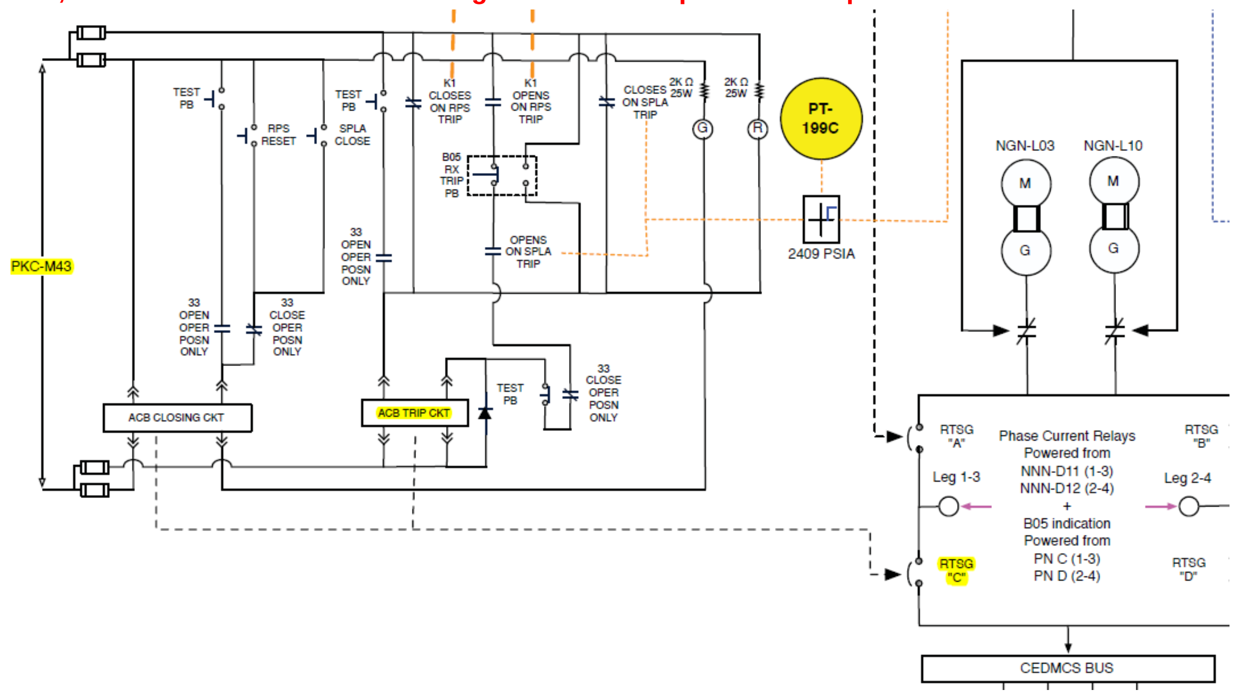
INSTRUCTIONS	CONTINGENCY ACTIONS
___ 5. Determine the effects of the de-energized bus. REFER TO Appendix E, Effects of the Loss of Channel C.	

Loss of PKC-M43 (125Vdc) causes only C RTSG breaker to open automatically.
Loss of PNC-D27 (120Vac) causes both A and C RTSG breakers to open automatically.

Appendix E, Effects of the Loss of Channel C

System	PKC M43	PKC D23	PNC D27	Response
SB	X			RTSG Breaker C trips open due to the UV relay de-energizing.
			X	RTSG Breaker A and C trip open due to loss of power to one leg of the RPS logic matrices BC, BD, CD. RTSG Breaker C trips on a SPLA trip and loss of power to RPS Initiation path #3. Lose power to all Channel C input parameter instruments resulting in 1-3 half leg trips on all parameters that have a trip setpoint. Parameters that fail high or low are inoperable. CEAC 2 in all CPC channels becomes inop due to loss of power to RSPTs and may generate penalty factors when re-energized.

PKC-M43 supplies control power to the C RTSG breaker control circuit. When control power is lost, the C RTSG breaker Undervoltage Coil will lose power and trip the breaker.



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Containment Spray: Ability to manually operate and/or monitor in the control room: Containment spray reset switches	Tier	2		
	Group	1		
	K/A	026 A4.05		
	IR	3.5		

Question 53

Given the following conditions:

- Unit 2 is operating at 100% power
- An inadvertent Train ‘A’ CSAS has occurred
- The CRS has entered 40AO-9ZZ17, Inadvertent PPS-ESFAS Actuations
- The crew has closed the ‘A’ CS Header Isolation Valve, SIA-UV-672, and stopped ‘A’ CS Pump, SIA-P03, per the AOP

Following the listed manipulations, the OVERRIDE light will be illuminated on ‘A’ CS Header Isolation Valve ___(1)___ , and AFTER the crew has reset the CSAS signal (but taken no further action) the OVERRIDE light(s) will ___(2)___ .

A. (1) ONLY
(2) be extinguished

B. (1) ONLY
(2) remain illuminated

C. (1) AND ‘A’ CS Pump
(2) be extinguished

D. (1) AND ‘A’ CS Pump
(2) remain illuminated

Proposed Answer:	A

Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible since there are components which must have control power cycled in order to restore normal functionality following an inadvertent CSAS (i.e. the CS Pump), however that is only necessary prior to resetting the CSAS signal.
C.	First part is plausible since this would be true if a SIAS had occurred, however the OVERRIDE light for the CS Pump is not enabled if the pump was started as a result of a CSAS with no SIAS signal. Second part is correct.
D.	First part is plausible since this would be true if a SIAS had occurred, however the OVERRIDE light for the CS Pump is not enabled if the pump was started as a result of a CSAS with no SIAS signal. Second part is plausible since there are components which must have control power cycled in order to restore normal functionality following an inadvertent CSAS (i.e. the CS Pump), however that is only necessary prior to resetting the CSAS signal.

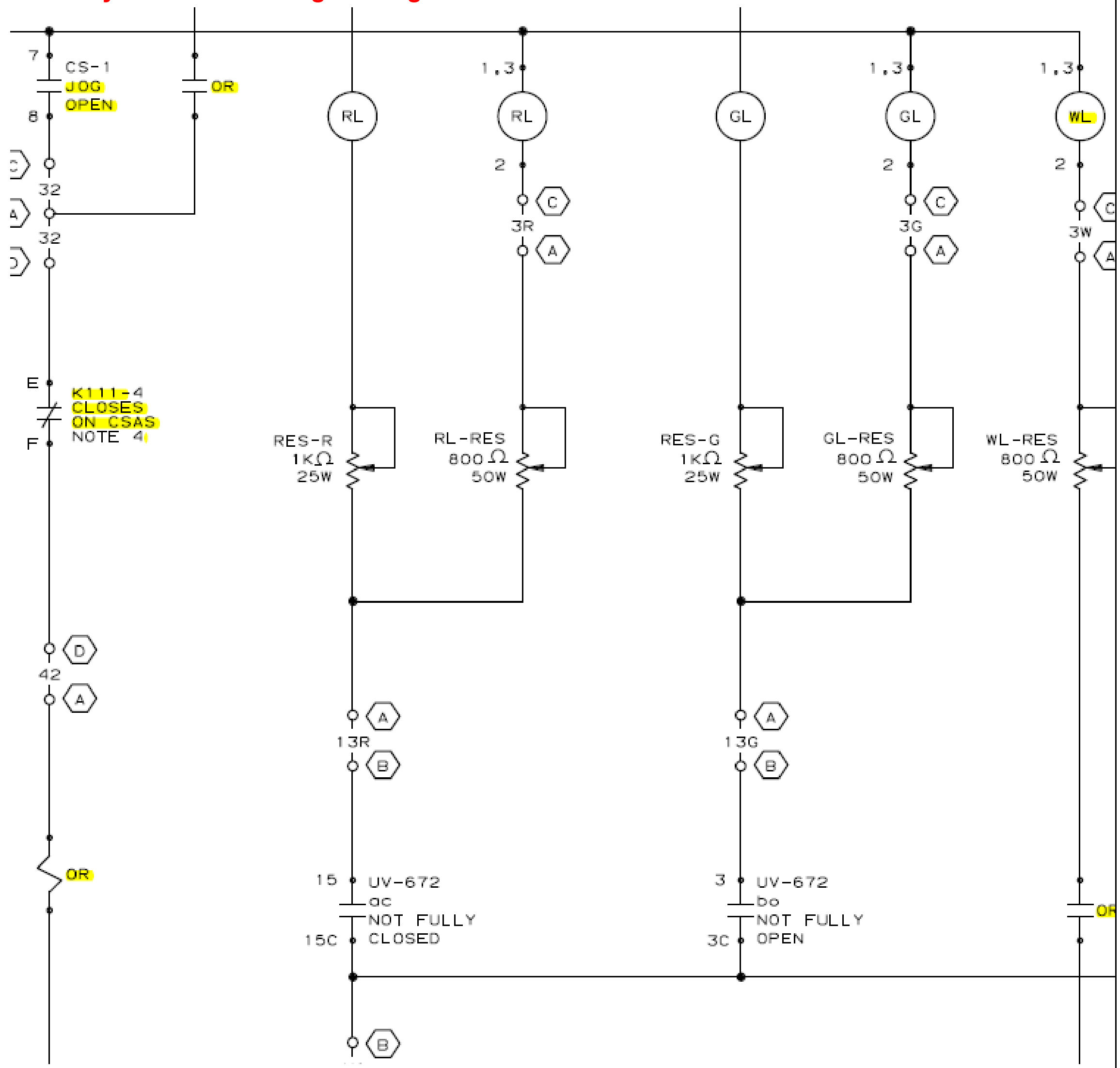
Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	7	
Reference Provided:	N	
Learning Objective:	Determine the impact of an inadvertent CSAS actuation and the actions needed to restore Plant stability.	

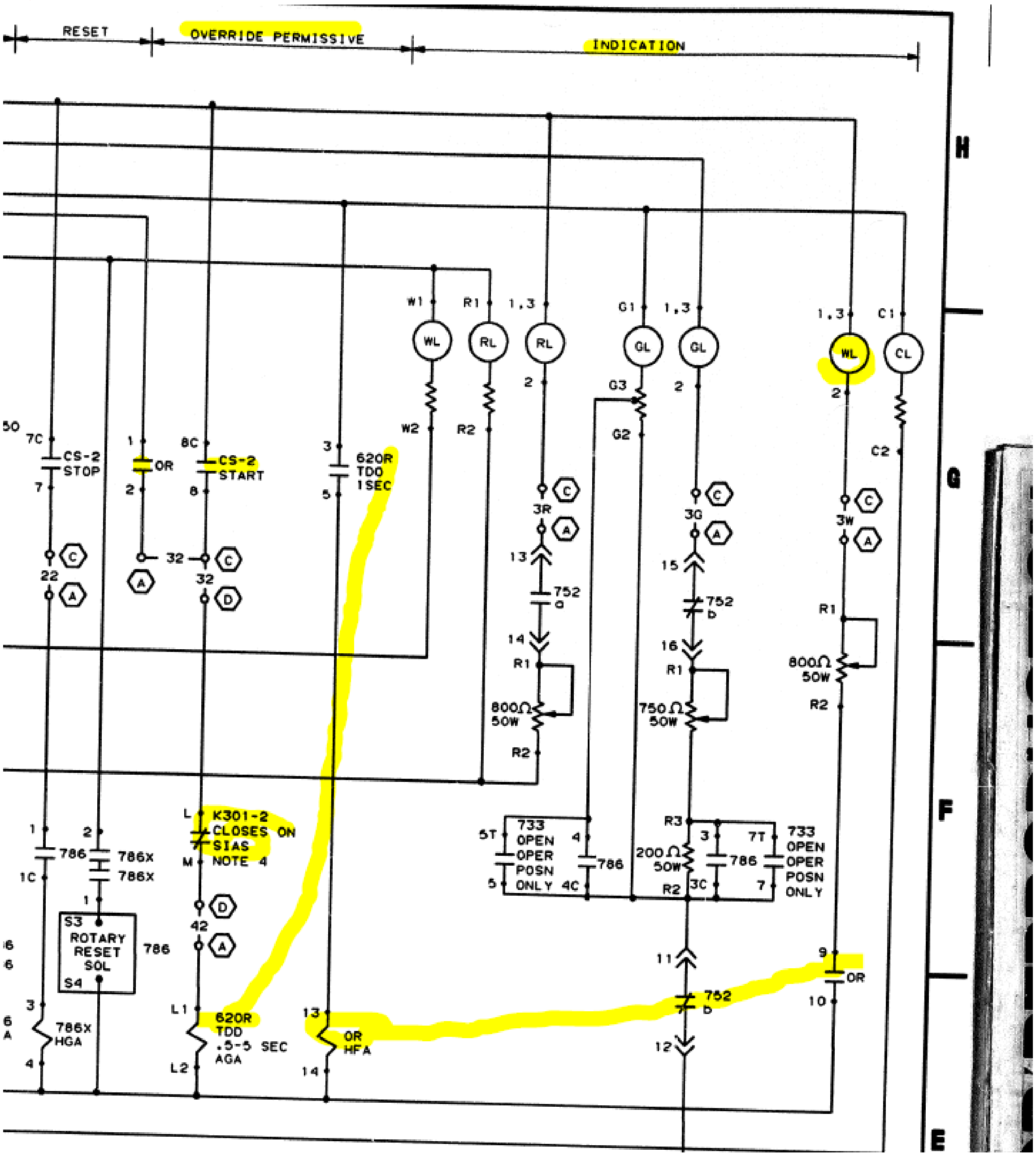
Technical Reference:	**Drawing 02-E-SIB-020 (Schematic for CS MOV SIA-UV-672)**

SIA-UV-672 auto opens on a CSAS signal. If CS-1 (MCR HS) is taken to Jog Open, the OR relay will energize. This will close the associated OR contacts. One will energize WL (override white light). The OR relay (override relay) will remain energized until the CSAS signal is reset. Once CSAS is reset, contact K111-4 will open. This will de-energize the OR relay and the white light will go dark.



Technical Reference:	Drawing 02-E-SIB-003 (CS Pump schematic)

CS Pump override operation differs from the CS MOV. The override circuit is enabled by a SIAS signal and NOT a CSAS signal as shown below. When the pump hand switch is held in STOP, the breaker is tripped and anti-pumped. The white override light will not be lit because a SIAS signal is not present.



Technical Reference:	40AO-9ZZ17 Inadvertent PPS-ESFAS Actuations

5.0 CSAS

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 3. **IF BOTH** of the following: • Any Containment Spray Pump is running • The running Containment Spray Pump is **NOT** being used for SDC **THEN** perform the following: a. **IF** SIAS has **NOT** actuated, **THEN** place the Containment Spray Pump hand switch in "STOP" to anti-pump the CS Pump. b. **IF** SIAS has actuated, **THEN** override and stop the Containment Spray Pump.	
____ 4. Override and close all open Containment Spray Header Isolation Valves.	____4.1 **IF BOTH** of the following: • Train A CSAS has actuated • SIA-UV-672, CS A Discharge
____ 14. **IF** a CS Pump needs to be started, **THEN** perform the following: a. **IF** SIAS has **NOT** actuated, **THEN** perform the following: 1) Inform an operator that the CS Pump breaker will close upon restoration of control power. 2) Direct the operator to cycle control power to the CS Pump breaker(s). b. **IF** SIAS actuated while the CS Pumps were stopped, **THEN** perform the following: 1) Place the CS Pump handswitch to "START" and release the switch. 2) Place the CS Pump handswitch to "START".	

Appendix C, PPS-ESFAS Check

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

If the BOP ESFAS Load Sequencer is in mode 4, override lights for sequenced loads will not clear until the DG is shutdown.

____ 4. **IF** the actuation has been reset, **THEN** perform the following in the appropriate attachment(s) of this appendix to ensure actuated components are aligned as desired:

a. Ensure that component status is appropriate for current plant conditions.

b. Circle the as left condition of all components.

c. Check that all components have power available.

d. Check that the white override light is extinguished for all components.

e. Check that all SESS alarms are clear for all components.

f. Inform the CRS of any discrepancies.

____ 4.1 **IF** the actuation will **NOT** be reset, **THEN** perform the following:

a. **IF BOTH** of the following:

- Any DG(s) is running due to AFAS or SIAS
- DG operation is **NOT** needed

THEN PERFORM **ANY** of the following to shutdown the Diesel Generator(s):

- 40OP-9DG01, Emergency Diesel Generator A
- 40OP-9DG02, Emergency Diesel Generator B

b. Override and align equipment as directed by the CRS.

White override lights should be extinguished after the CSAS actuation has been reset.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Main and Reheat Steam: Ability to manually operate and/or monitor in the control room: Main steam supply valves	Tier	2		
	Group	1		
	K/A	039 A4.01		
	IR	2.9		

Question 54

Given the following conditions:

- Unit 1 is preparing to place the Main Steam Lines in service with Condenser vacuum established per 40OP-9SG01, Main Steam

To open SGE-UV-169, SG 1 MSIV Bypass Valve, the BOP will place ___(1)___ of the two MSIV Bypass Valve handswitch(s) in OPEN.

After D/P across Line 1 MSIV UV-170 has been equalized, the BOP will place ___(2)___ of the two MSIV handswitches in SLOW OPEN.

A. (1) BOTH
(2) BOTH

B. (1) BOTH
(2) ONLY one

C. (1) ONLY one
(2) BOTH

D. (1) ONLY one
(2) ONLY one

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible since the MSIV Bypass Valves are opened using both handswitches, however the MSIVs are opened using only one handswitch.
B.	Correct.
C.	First part is plausible since MSIVs are opened using only one handswitch, however MSIV Bypass Valves are opened using both handswitches. Second part is plausible since the MSIV Bypass Valves are opened using both handswitches, however the MSIVs are opened using only one handswitch.
D.	First part is plausible since MSIVs are opened using only one handswitch, however MSIV Bypass Valves are opened using both handswitches. Second part is correct.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	4	
Reference Provided:	N	
Learning Objective:	Explain the operation of the Main Steam Isolation Valves under normal operating conditions.	

Technical Reference:	40OP-9SG01 Main Steam

Both handswitches must be placed in OPEN to open a MSIV Bypass (equalizing) valve. However, if either handswitch is taken to CLOSE, the MSIV Bypass valve will close.

Only one handswitch has to be placed in SLOW OPEN to open an MSIV.

6.1 Placing the Main Steam Lines in Service With MSIVs Closed and Condenser Vacuum Established

6.1.18 **IF** SGE-UV-169, Main Steam Equalizing Valve, is CLOSED,
THEN take BOTH of the following handswitches to OPEN:

___ • SGA-HS-169A, MSIV BYPASS ISOL VLV UY-169A

___ • SGB-HS-169B, MSIV BYPASS ISOL VLV UY-169B

NOTE

___ Due to limit switch arrangement, stroking an MSIV with one hydraulic train may give intermediate indication on the inactive hydraulic system indicators.

_ 6.1.28 Perform the following to OPEN all of the MSIVs:

___ A. **IF** SGE-UV-170, SG1 Line 1 Main Steam Isolation Valve, is to be OPENED,
THEN perform the following:

___ 1. Take BOTH of the following SG1 handswitches to SLOW CLOSE, to reset the valve logic:

___ • SGA-HS-170A, LINE 1 MSIV UV-170

___ • SGB-HS-170B, LINE 1 MSIV UV-170

___ 2. Take ONE of the following handswitches to SLOW OPEN:

___ • SGA-HS-170A, LINE 1 MSIV UV-170

___ • SGB-HS-170B, LINE 1 MSIV UV-170

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Service Water: Ability to determine operability and/or availability of safety related equipment	Tier	2		
	Group	1		
	K/A	076 G 2.2.37		
	IR	3.6		

Question 55

Regarding the ‘A’ Spray Pond Pump:

(1) If the pump is running due to an AUTO-start, the Control Board indication will show a…

(2) If the pump failed to AUTO-start and CANNOT be manually started, the Control Room indication will show…

A. (1) green light with a red-flagged handswitch
(2) white SESS alarm ONLY

B. (1) green light with a red-flagged handswitch
(2) blue AND white SESS alarms

C. (1) red light with a green-flagged handswitch
(2) white SESS alarm ONLY

D. (1) red light with a green-flagged handswitch
(2) blue AND white SESS alarms

Proposed Answer:	**D**

Explanations:	
A.	First part is plausible since there will be differing light and handswitch indications, however the reverse is correct. Second part is correct since the white SESS light is indicative of an overcurrent trip on the pump breaker, however the blue alarm is indicative of a failure to go to it's actuated position.
B.	First part is plausible since there will be differing light and handswitch indications, however the reverse is correct. Second part is correct.
C.	First part is correct. Second part is correct since the white SESS light is indicative of an overcurrent trip on the pump breaker, however the blue alarm is indicative of a failure to go to its actuated position.
D.	Correct.

Question Source:	**x**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		**Memory or Fundamental Knowledge**
	x	**Comprehension or Analysis**

Level of Difficultly:	**3**	
10CFR55.41:	**7**	
Reference Provided:	**N**	
Learning Objective:	**Explain the operation of the Safety Equipment Status System system level windows and component level windows under normal operating conditions.**	

Technical Reference:	Class 1E Distribution (PB/PG/PH) System Lesson Plan

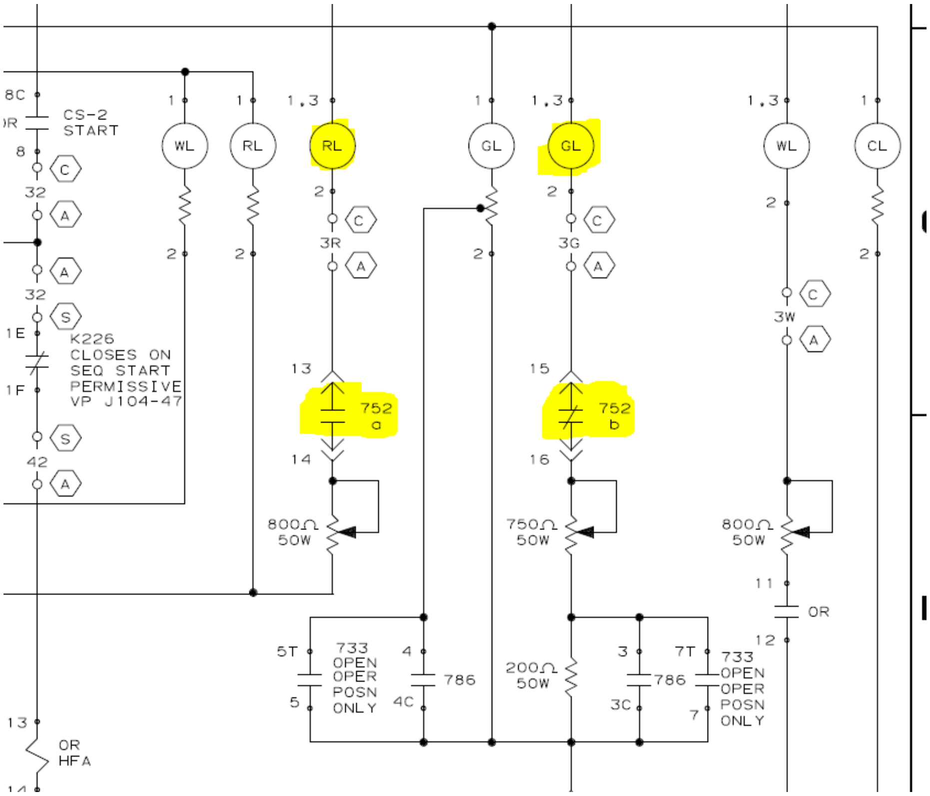
Light	Purpose
Red	Breaker CLOSED Continuity through breaker trip coil. If the red light is not lit, the trip circuit may not function. **(CRAI 3154411).** *Note: The current through the red light is limited by a resistor. This resistor keeps the current to a value that will illuminate the light but not generate enough magnetic field to activate the breaker's trip coil.*
Green	Breaker OPEN *Note: Controlled by breaker auxiliary contacts with a current limiting resistor is in series with the light.*

If the pump is auto started, the flag on the MCR HS will remain "green flagged". A MCR manual start would cause a "red flag" on the handswitch.

Technical Reference:	Drawing 1M-SPA-P01 (Spray Pond Pump A schematic)

Control circuit for Spray Pond Pump breaker (shown de-energized with the breaker open).

The red light would be lit if the breaker is closed (breaker closed causes 752a contact to close).
Green light would be dark if the breaker is closed (752b contact would be open)



Technical Reference:	40AL-9ES2A Safety Equipment Status System Panel ESA-UA-2A Alarm Responses

Blue SESS light lit if an auto start signal exists and the pump didn't start.

Response Section

Essential Spray Pond Pump A SPA-P01

SEAS 10J
ESS
SPRAY POND
PMP A
P01

Point ID	Description	Setpoint
PBA-S03C-752 contact	Ckt Bkr for SPA-P01 "A" Spray Pond Pump	SPA-P01 NOT running

White SESS light lit if a failure in the breaker control circuit has occurred that will prevent the pump from starting (automatically or manually from the MCR).

Response Section

Essential Spray Pond Pump A SPA-P01

SEIS 10J
ESS
SPRAY POND
PMP A
P01

Point ID	Description	Setpoint
PBA-S03C-786 Relay	Lock-Out Relay for ESS Spray Pond Pump A	786 Relay tripped
OR		
PBA-S03C-762C Relay contact	Agastat TD Relay for ESS Spray Pond Pump A (762C)	PBA-S03C Control power loss, NOT racked to the operate position, closing springs NOT charged
OR		
PBA-S03C-762T Relay contact	Agastat TD Relay for ESS Spray Pond Pump A (762T)	PBA-S03C Control power loss, NOT racked to the operate position

NOTE

___ This status light active with a concurrent SIAS, CREFAS, or CRVIAS signal present will result in blue Train A SEAS 10J, ESS SPRAY POND PMP A P01 in alarm.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Control Rod Drive: Knowledge of the physical connections and/or cause-effect relationships between the CRDS and the following systems: NIS and RPS	**Tier**	**2**		
	Group	**2**		
	K/A	**001 K1.05**		
	IR	**4.5**		

Question 56

(1) The input signal for the HI LOG POWER trip comes from the…

(2) The input signal for the VOPT trip comes from the…

A. (1) the middle safety channel NI ONLY
   (2) the average of all 3 safety channel NIs

B. (1) the middle safety channel NI ONLY
   (2) the highest indicated power of the 3 safety channel NIs

C. (1) the average of all 3 safety channel NIs
   (2) the average of all 3 safety channel NIs

D. (1) the average of all 3 safety channel NIs
   (2) the highest indicated power of the 3 safety channel NIs

Proposed Answer:	**A**

Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible since there are other systems with multiple inputs which use a “high select” such as DFWCS, and using the highest indicated power would be the most conservative assessment of power, however VOPT is triggered from the average of all 3 safety channel NIs.
C.	First part is plausible since VOPT uses the average of all 3 safety channel NIs, however only the middle detector is used for log power and thus only the middle detector is used for the hi log power trip. Second part is correct.
D.	First part is plausible since VOPT uses the average of all 3 safety channel NIs, however only the middle detector is used for log power and thus only the middle detector is used for the hi log power trip. Second part is plausible since there are other systems with multiple inputs which use a “high select” such as DFWCS, and using the highest indicated power would be the most conservative assessment of power, however VOPT is triggered from the average of all 3 safety channel NIs.

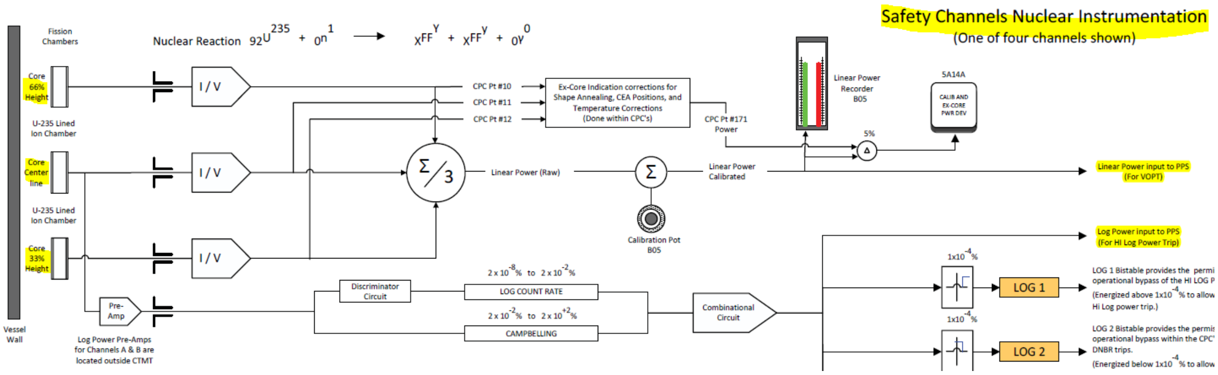
Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	**X**	**Memory or Fundamental Knowledge**
		Comprehension or Analysis

Level of Difficultly:	**3**	
10CFR55.41:	**3**	
Reference Provided:	**N**	
Learning Objective:	**Describe the circuitry associated with the Safety Channel Nuclear Instruments.**	

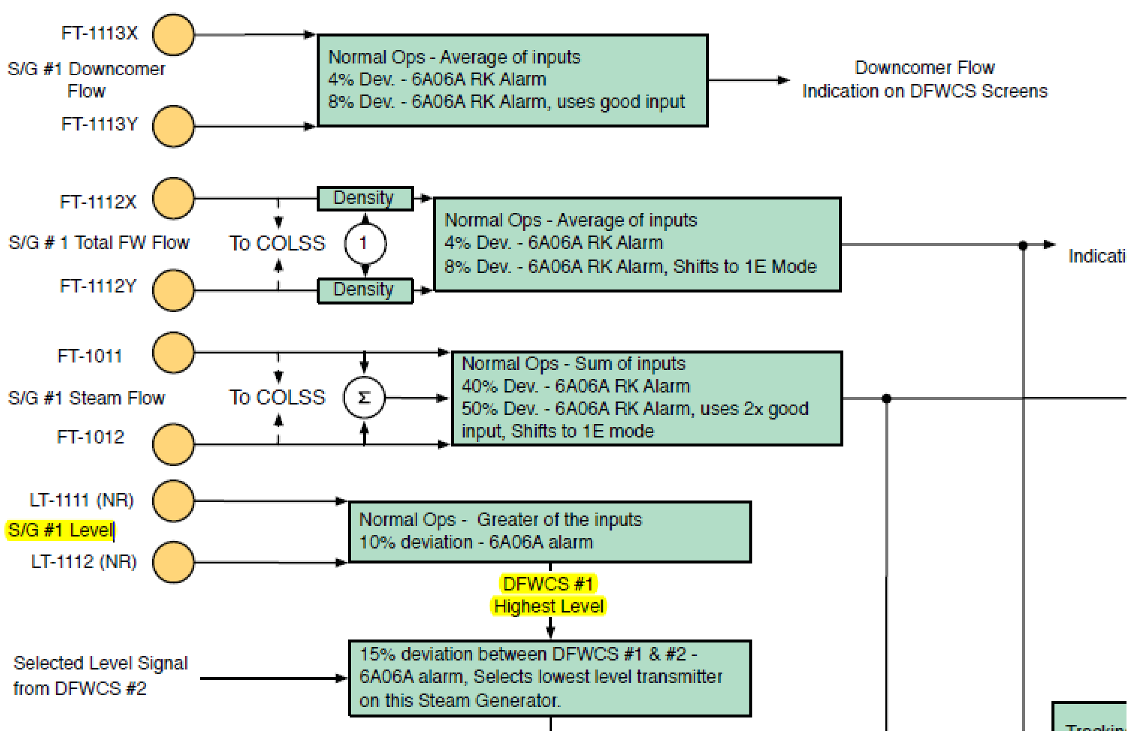
All three NIs (upper, middle, & lower) are averaged together for the input into the VOPT reactor trip.

The middle detector output is sent to the Hi Log power reactor trip.



DFWCS uses the higher of the two SG water level inputs for controlling SGWL

DFWCS - Digital Feedwater Control System (3 Element)



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Reactor Coolant: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCS controls including: Temperature	Tier	2		
	Group	2		
	K/A	002 A1.03		
	IR	3.7		

Question 57

(1) During a power ascension from 20% to 100% power, if RCS Tcold is kept on program, Tcold will _____ during the power ascension

(2) To ensure LCO 3.4.2, RCS Minimum Temperature for Criticality, remains met, RCS Tcold must remain greater than or equal to a MINIMUM of _____ .

A. (1) rise
(2) 545°F

B. (1) rise
(2) 550°F

C. (1) lower
(2) 545°F

D. (1) lower
(2) 550°F

Proposed Answer:	C

Explanations:	
A.	First part is plausible since Thot and Tavg will both rise, however Tcold will lower during a power ascension. Second part is correct.
B.	First part is plausible since Thot and Tavg will both rise, however Tcold will lower during a power ascension. Second part is plausible as 550°F is the minimum Tcold to remain in the “acceptable area of operation” per LCO 3.4.1, however the minimum RCS Tcold for criticality is 545°F..
C.	Correct.
D.	First part is correct. Second part is plausible as 550°F is the minimum Tcold to remain in the “acceptable area of operation” per LCO 3.4.1, however the minimum RCS Tcold for criticality is 545°F..

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	2	
Reference Provided:	N	
Learning Objective:	Describe LCO 3.4.2, Minimum Temperature for Criticality, including the basis.	

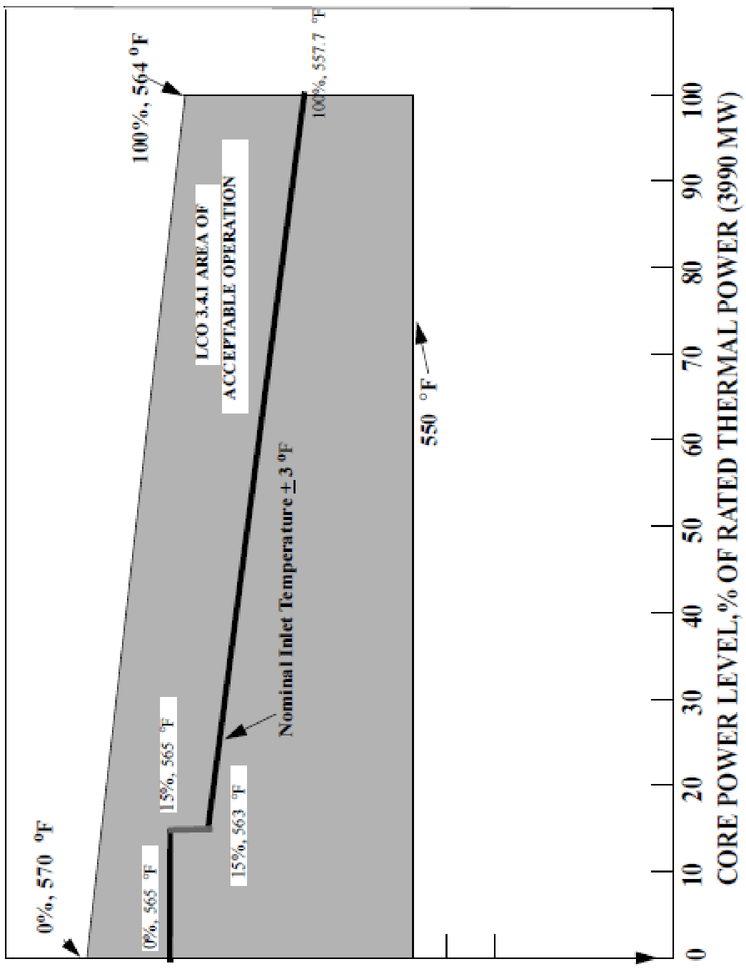
Technical Reference:	40OP-9ZZ05 Power Operations
The GOPs direct maintaining RCS temps on program per the CDB.	
___ 6.2.23 Maintain BOTH of the following RCS parameters:	
___ • RCS T_{cold} at or near the T_{cold} program found in the Core Data Book	
___ • RCS T_{avg} within 3°F of T_{ref} by adjusting RCS boron concentration	

Technical Reference:	Core Data Book

Notes 1.0

Variable T_{cold} Program

Reactor Coolant Cold Leg Temperature Vs. Core Power Level



Not to Scale

COLD LEG TEMPERATURE, °F

Technical Reference:	Technical Specifications

545°F is the minimum temperature for criticality

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each RCS loop temperature (T_{cold}) shall be ≥ 545°F.

APPLICABILITY: MODE 1,
MODE 2 with K_{eff} ≥ 1.0.

550°F is the minimum temperature for the LCO 3.4.1 DNB

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, cold leg temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ≥ 2130 psia and ≤ 2295 psia; and

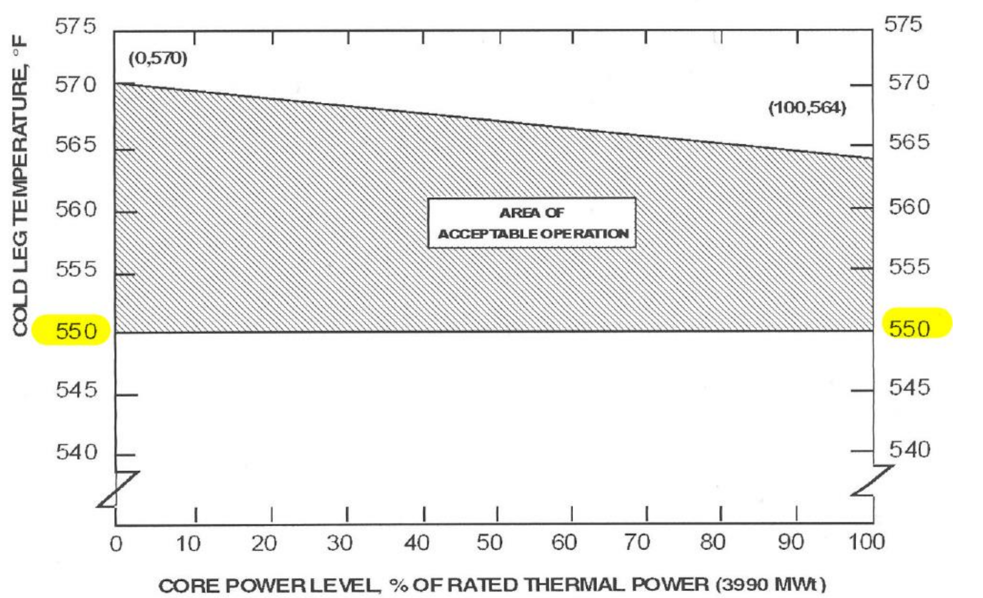
b. RCS cold leg temperature (T_c) shall be within the area of acceptable operation shown in Figure 3.4.1-1; and

c. RCS total flow rate ≥ 155.8 E6 lbm/hour.

APPLICABILITY: MODE 1 for RCS total flow rate,

MODES 1 and 2 for pressurizer pressure,

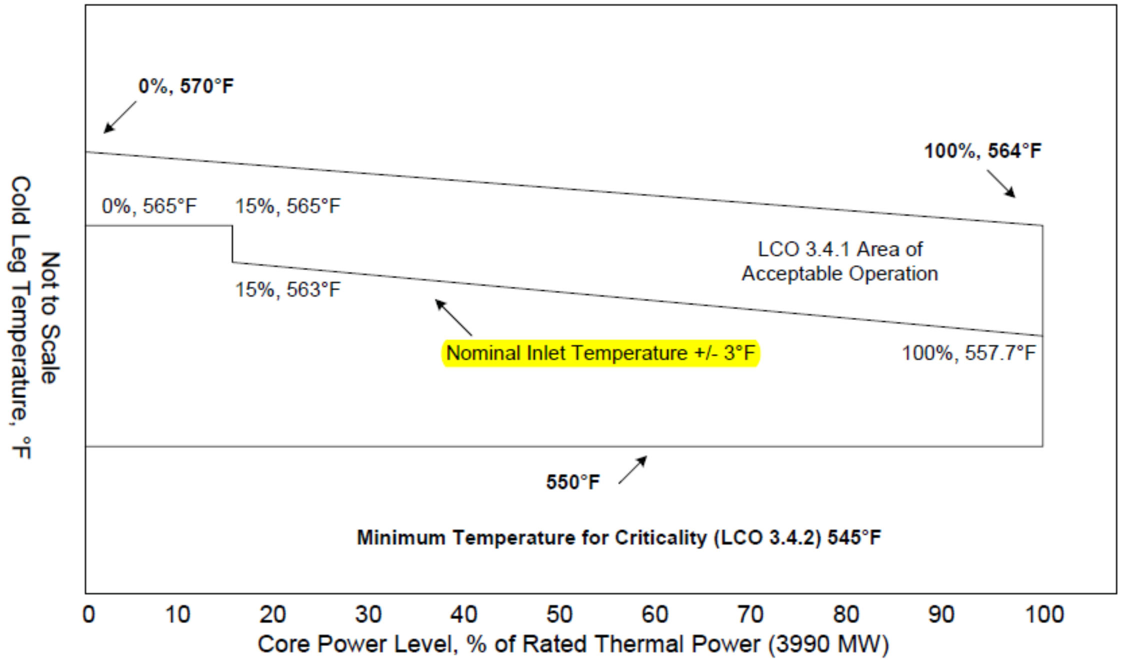
MODE 1 for RCS cold leg temperature (Tc).



Technical Reference:	PVGS Operator Information Manual

The "nominal inlet temperature" is showing programmed Tcold from 0% to 100% RTP.

Reactor Coolant Cold Leg Temperature Vs. Core Power Level Unit 1



Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Hydrogen Recombiner and Purge Control: Knowledge of the operational implications of the following concepts as they apply to the HRPS: Sources of hydrogen within containment	Tier	2	[redacted]	
	Group	2	[redacted]	
	K/A	028 K5.03		
	IR	2.9	[redacted]	

Question 58

During a LOCA, the PRIMARY source of hydrogen production during core uncovery comes from the ___(1)___ .

Per 40DP-9ZZ04, Time Critical Action Program, the Hydrogen Analyzers must be placed in service within a MAXIMUM of ___(2)___ from the start of the LOCA.

A. (1) radiolysis of water
(2) 15 minutes

B. (1) radiolysis of water
(2) 30 minutes

C. (1) zirconium-steam reaction
(2) 15 minutes

D. (1) zirconium-steam reaction
(2) 30 minutes

Proposed Answer:	D

Explanations:	
A.	First part is plausible as this is a contributor to the hydrogen generated following a LOCA, however the primary source is from the zirc-steam reaction. Second part is plausible since the safety function status checks are required to be performed every 15 minutes while operating in the EOPs, however the time limit to get a hydrogen analyzer in service is 30 minutes.
B.	First part is plausible as this is a contributor to the hydrogen generated following a LOCA, however the primary source is from the zirc-steam reaction. Second part is correct.
C.	First part is correct. Second part is plausible since the safety function status checks are required to be performed every 15 minutes while operating in the EOPs, however the time limit to get a hydrogen analyzer in service is 30 minutes.
D.	Correct.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	7	
Reference Provided:	N	
Learning Objective:	Given conditions of a LOCA, state the time limitation associated with placing the Hydrogen Analyzers in service in accordance with 40EP-9EO03.	

Technical Reference:	Containment Environmental Effects Lesson Plan

Zirconium

The Zirconium-Steam reaction is the primary hydrogen production method inside the vessel during core uncovery conditions. It can generate large amounts of hydrogen due to large quantity of zirconium in the core. The reaction is exothermic (generates heat) and is temperature dependent. At increased temperature levels, the reaction can become self sustaining and heat produced can be more than from decay heat.

Per 10 CFR 50:

50.46 Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.

(b)(1) *Peak cladding temperature*. The calculated maximum fuel element cladding temperature shall not exceed 2200º F.

3) *Maximum hydrogen generation*. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

Radiolysis

This is the separation of water into its constituent components under the influence of a radiation field. Radiation involved includes alpha, beta, gamma and neutrons. Hydrogen production by this method is of concern for extended periods after significant core damage has occurred and fuel and coolant are mixed together.

Technical Reference:	40DP-9ZZ04 Time Critical Action (TCA) Program

TCA	Action	Time Limit	Time Zero	Validation Method	Procedure	Org	Source Document (other info)
TCA-55	Place H2 Analyzers in service following LOCA	30 minutes	LOCA event	Simulator	40EP-9EO03 40EP-9EO05 40EP-9EO09	Ops	UFSAR 6.2.5.2.1

Technical Reference:	40DP-9AP16 Emergency Operating Procedure Users Guide

4.18.3 Safety Function Status Checks

A. The review and assessment of the SFSC is the responsibility of the CRS. This will normally be assigned to the STA.

B. Safety Function Status should be checked approximately every 15 minutes while an ORP, FRP, or LMFR is in use.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Spent Fuel Pool Cooling: Knowledge of the effect that a loss or malfunction of the Spent Fuel Pool Cooling System will have on the following: Area and ventilation radiation monitoring systems	Tier	2		
	Group	2		
	K/A	033 K3.02		
	IR	2.8		

Question 59

Given the following conditions:

- A leak has occurred in the Spent Fuel Pool
- Radiation levels are rising in the area
- RU-31, Spent Fuel Pool Area monitor is in HIGH alarm
- RU-145, Fuel Building Ventilation Low Range Gas monitor is in ALERT alarm

Which ONE of the following describes the plant response to the listed conditions?

A. Both FBEVAS and CREFAS actuated DIRECTLY from the HIGH alarm on RU-31

B. FBEVAS actuated on RU-31 HIGH alarm, and CREFAS actuated on cross-trip

C. CREFAS actuated on RU-31 HIGH alarm, and FBEVAS actuated on cross-trip

D. FBEVAS actuated on RU-31 HIGH alarm, CREFAS actuated on the RU-145 ALERT alarm

Proposed Answer:	B

Explanations:	
A.	Plausible if assumed that both CREFAS and FBEVAS are actuated from a high alarm on RU-31, however the CREFAS is actuated via a cross-trip
B.	Correct.
C.	Plausible if assumed that RU-31 will actuated CREFAS directly and cross-trip FBEVAS, however the opposite is correct.
D.	First half of the answer is correct, and the second half is plausible if thought that RU-145 will actuate CREFAS at the alert level.

Question Source:		New	
	x	Bank	
		Modified	
	x	Previous NRC Exam	2018 NRC Q65

Cognitive Level:		Memory or Fundamental Knowledge
	x	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	11	
Reference Provided:	N	
Learning Objective:	Explain the operation of the FBEVAS Module.	

Technical Reference:	74AL-9SQ01 Radiation Monitoring System Alarm Validation and Response

Response Section

OPS RU-31
HIGH
ALERT

Monitor Name: SPENT FUEL POOL AREA

Location: 140-foot Fuel Building, East wall

Point ID	Description	Setpoint
N/A	Channel 1	N/A

AUTOMATIC ACTION

- HIGH alarm on RU-31 initiates a Fuel Building Essential Ventilation Actuation Signal (FBEVAS) and cross-trips a Control Room Essential Filtration Actuation Signal (CREFAS).

MANUAL ACTIONS

___ 1. **IF** a HIGH alarm is received on RU-31, **THEN** perform the following:

____ 1.1 Perform BOTH of the following:

___ 1.1.1 Check FBEVAS has actuated.

___ 1.1.2 Check CREFAS cross-tripped.

Response Section

OPS RU-145 Gas Channel
HIGH
ALERT

Monitor Name:

FUEL BUILDING VENTILATION (LOW RANGE GAS)

Location:

Detector - 176-foot Fuel Building, West End

Microcomputer - 120-foot Control Building, West Wall

Point ID	Description	Setpoint
N/A	Channel 1	N/A

AUTOMATIC ACTION

- HIGH alarm initiates a Fuel Building Essential Ventilation Actuation Signal (FBEVAS)

MANUAL ACTIONS

___ 1. **IF** a HIGH alarm is received on RU-145, **THEN** perform ONE of the following:

____ 1.1 Check BOTH of the following:

___ • FBEVAS actuated

___ • FBEVAS actuation cross-tripped Control Room Essential Filtration Actuation Signal (CREFAS).

Technical Reference:	40AL-9RK5A Panel B05A Alarm Responses

Response Section

5A04A

CREFAS A

Control Room Essential Filtration Actuation Signal Train A

Point ID	Description	Setpoint
SAYS5	Control Room Ess Filtration Actuation Signal A (RU-29)	equal to or less than 2×10^{-5} µci/cm^3 or FBEVAS or CPIAS

PROBABLE CAUSES

- Manual actuation from panel B05, SAA-HS-27 or A BOP-ESFAS panel test pushbutton
- SQA-RU-29, Control Room Ventilation Intake - Train A air monitor reading above setpoint
- FBEVAS Train A or B actuated
- CPIAS Train A or B actuated
- CREFAS Train B actuated
- SQA-RU-29, Control Room Ventilation Intake - Train A air monitor failure

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Steam Generator: Ability to (a) predict the impacts of the following malfunctions or operations on the SG; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Small break LOCA	Tier	2		
	Group	2		
	K/A	035 A2.06		
	IR	4.5		

Question 60

Given the following conditions:

- Unit 2 was tripped from 100% power due to a small break LOCA inside Containment
- SIAS and CIAS were manually actuated after the Reactor trip
- Current plant conditions are as follows:
    - Pressurizer level is 10% and slowly lowering
    - Pressurizer pressure is 1950 psia and slowly lowering
    - SG levels are both 40% NR and slowly lowering
    - SG pressures are both 1050 psia and slowly rising
    - Containment pressure is 3.2 psig and slowly rising
- The BOP is preparing to report the status of RCS Heat Removal

Assuming NO operator actions have been taken on Board 6, which of the following describes what the BOP should do per 40EP-9EO01, Standard Post Trip Actions and PVNGS EOP Operations Expectations?

Inform the CRS that they are going to ___(1)___ to control SG pressure and transition to ___(2)___ to control SG levels.

A. (1) transition to ADVs
(2) AFB-P01

B. (1) transition to ADVs
(2) AFN-P01

C. (1) take manual control of SBCS Valves 1007 and 1008
(2) AFB-P01

D. (1) take manual control of SBCS Valves 1007 and 1008
(2) AFN-P01

Proposed Answer:	A

Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible since AFN-P01 is normally preferred, however since SIAS actuated, AFB-P01 is already running and if an AFW Pump has received an auto start signal and is already running, it becomes the preferred pump to transition to.
C.	First part is plausible since ADVs are a last resort and 1007 and 1008 do not relieve to the Main Condenser, however since Containment pressure is > 3.0 psig, MSIS has actuated rendering 1007 and 1008 unavailable. Second part is correct.
D.	First part is plausible since ADVs are a last resort and 1007 and 1008 do not relieve to the Main Condenser, however since Containment pressure is > 3.0 psig, MSIS has actuated rendering 1007 and 1008 unavailable. Second part is plausible since AFN-P01 is normally preferred, however since SIAS actuated, AFB-P01 is already running and if an AFW Pump has received an auto start signal and is already running, it becomes the preferred pump to transition to.

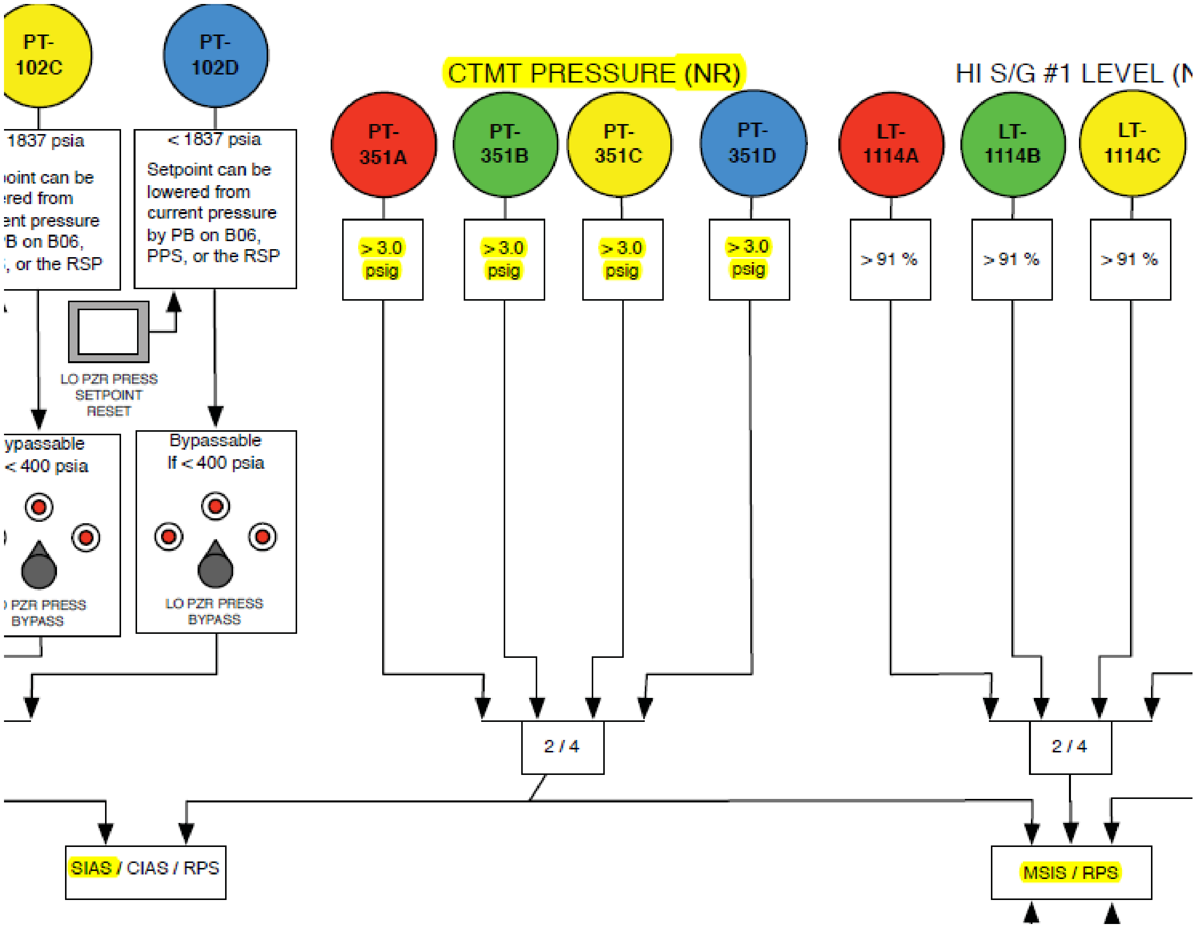
Question Source:	x	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		**Memory or Fundamental Knowledge**
	x	**Comprehension or Analysis**

Level of Difficultly:	3	
10CFR55.41:	8	
Reference Provided:	N	
Learning Objective:	**Given plant conditions following a reactor trip, analyze whether the RCS Heat Removal Safety Function is met and what contingency actions are required if it is not in accordance with 40EP-9EO01.**	

Technical Reference:	PVGS Operator Information Manual

SIAS and MSIS actuated if CTMT pressure is greater than 3 psig



Technical Reference:	EOP Operator Expectations

AFB auto starts from the SIAS signal. AFB is the least complex pump to use to start feeding.

4. The general priorities for restoration of feedwater to the steam generators are as follows:

   - Focus first on the systems and methods that are the least complex and most likely to be operational. Examples includes feed sources that are already in operation, such as using a MFP in manual or in reactor trip override mode, and the use of AFB after it has started post-SIAS actuation.

   - If an auxiliary feedwater pump must be started, then attempt to start AFN-P01, AFB-P01, or AFA-P01 (listed in order of preference except for LOOP events) from the Control Room. The operator must exercise caution when manually initiating feed or overriding AFAS actuated valves. Do not feed a steam generator with a DP lockout indicating a faulted steam generator.

MSIS isolates the main steam lines by closing the MSIVs, therefore SBCS is not available and the ADVs have to be used to control SG pressure.

7. If available, the SBCS should be used to control Tc. If the SBCS is not available, the ADVs should be used.

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Steam Dump/Turbine Bypass Control: Knowledge of the effect of a loss or malfunction on the following will have on the SDS: Controller and positioners, including ICS, S/G, CRDS	**Tier**	**2**	[redacted]	
	Group	**2**	[redacted]	
	K/A	**041 K6.03**		
	IR	**2.7**	[redacted]	

Question 61

Given the following conditions:

- Unit 2 is operating at 100% power
- RCN-HS-100, Pressure Control Channel X/Y Selector, is selected to Channel 'X'

Subsequently:

- RCN-PT-100X, Pressurizer Control Channel 'X', failed to 2500 psia

With NO operator action, the SBCS Auto Modulation Setpoint will ___(1)___ and the SBCS Auto Modulate Permissive light will be ___(2)___ .

A. (1) lower
   (2) illuminated

B. (1) lower
   (2) extinguished

C. (1) remain constant
   (2) illuminated

D. (1) remain constant
   (2) extinguished

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible as this would be the case if PT-100Y failed high, however the permissive light will not illuminate if Channel X fails high.
B.	Correct.
C.	First part is plausible since this would be the result of PT-100X failing low, however when it fails high, the modulation setpoint lowers. Second part is plausible as this would be the case if PT-100Y failed high, however the permissive light will not illuminate if Channel X fails high.
D.	First part is plausible since this would be the result of PT-100X failing low, however when it fails high, the modulation setpoint lowers. Second part is correct.

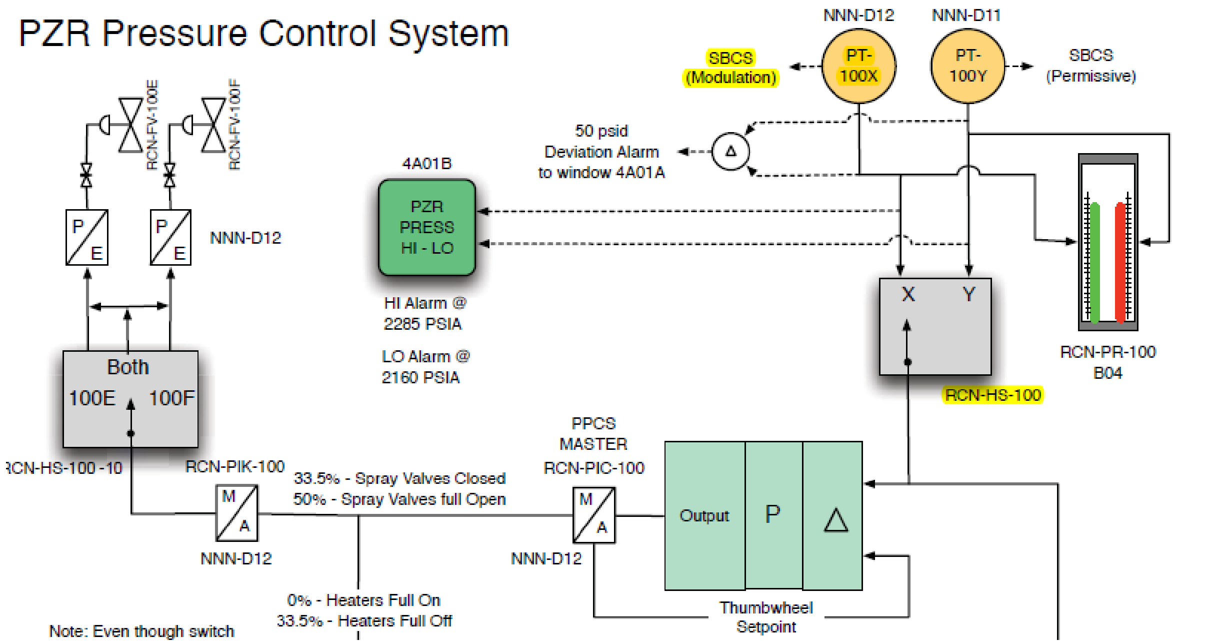
Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		**Memory or Fundamental Knowledge**
	X	**Comprehension or Analysis**

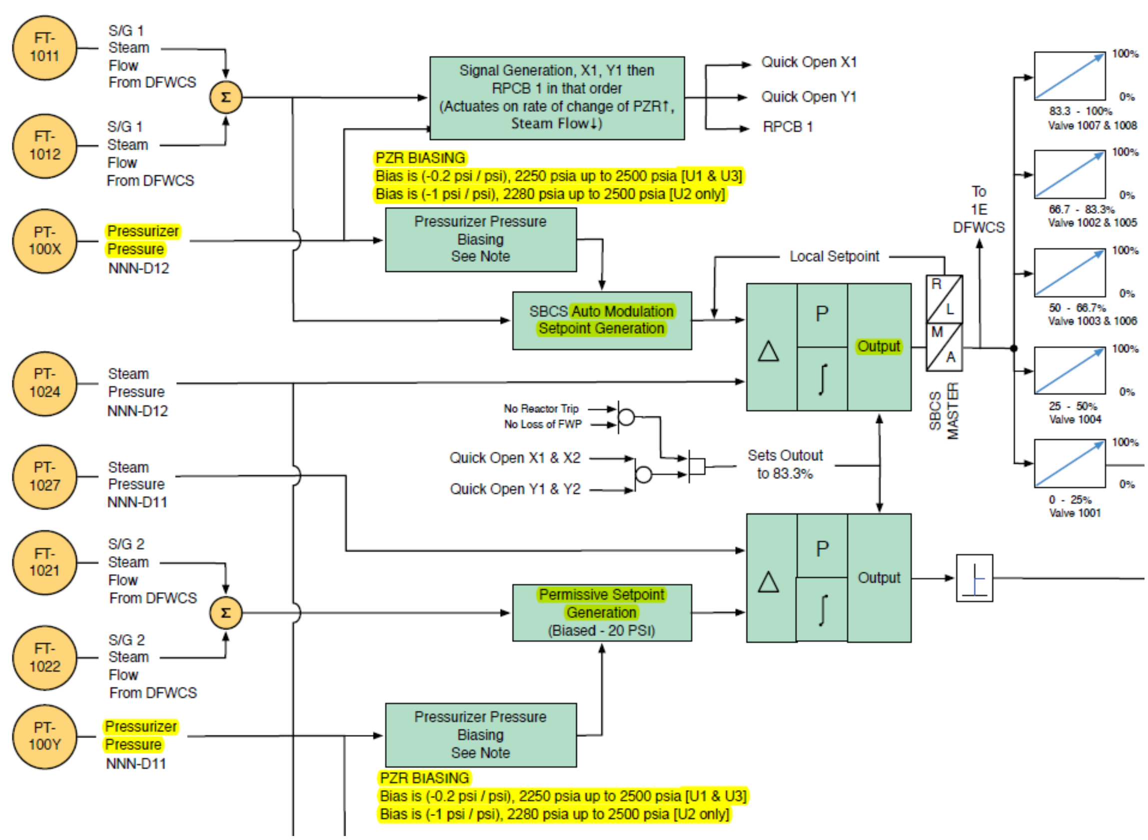
Level of Difficultly:	**3**
10CFR55.41:	**4**
Reference Provided:	**N**
Learning Objective:	**Describe the response of the SBCS to a failure of the following: Pressurizer Pressure**

PZR PT-100X inputs directly to the SBCS modulation controls.

PZR Pressure Control System



Steam Bypass Control System Functional



RCN-PT-100X	Pressurizer Pressure Transmitter 100X is used to BIAS(downward) the Auto Modulation Setpoint and also used in the Quick Open X1, Y1, and RPCB 1 generation.	HIGH	Effect on Quick Open 1. The failure HIGH will transiently cause the Quick Open or RPCB bistables to actuate. 2. The failure HIGH will prevent the system from seeing the PZR Pressure increase on a real Load Rejection. (delay Quick Opens) Effect on Auto Modulation Setpoint 1. The failure HIGH will BIAS the SBCS Auto Modulation Setpoint lower by up to 220 psia maximum. (Units 2 & 3, 50 psia for Unit 1) 2. The SBCS system may generate a Auto Modulation Signal due to real Steam Pressure being above setpoint. (100% modulation signal). 3. Elevated Tave condition (CEA drop) should be addressed in Local / Auto
		LOW	Effect on Quick Open 1. The failure LOW will not cause the Quick Open or RPCB bistables to actuate. (wrong direction) 2. The failure LOW will prevent the system from seeing the PZR Pressure increase on a real Load Rejection. (delay Quick Opens) Effect on Auto Modulation Setpoint 1. The failure LOW will not BIAS the SBCS Auto Modulation Setpoint lower. (wrong direction) 2. The SBCS system can still generate a Auto Modulation Signal when real Steam Pressure goes above setpoint.

RCN-PT-100Y	Pressurizer Pressure Transmitter 100Y is used to BIAS(downward) the Auto Permissive Setpoint and also used in the Quick Open X2, Y2, and RPCB 2 generation.	HIGH	Effect on Quick Open 1. The failure HIGH will transiently cause the Quick Open or RPCB bistables to actuate. 2. The failure HIGH will prevent the system from seeing the PZR Pressure increase on a real Load Rejection. (delay Quick Opens) Effect on Auto Permissive Setpoint 1. The failure HIGH will BIAS the SBCS Auto Permissive Setpoint lower by up to 220 psia maximum. (Units 2 & 3, 50 psia for Unit 1) 2. The SBCS system may generate a Auto Permissive Signal due to real Steam Pressure being above setpoint.
		LOW	Effect on Quick Open 1. The failure LOW will not cause the Quick Open or RPCB bistables to actuate. 2. The failure LOW will prevent the system from seeing the PZR Pressure increase on a real Load Rejection. (delay Quick Opens) Effect on Auto Permissive Setpoint 1. The failure LOW will not BIAS the SBCS Auto Permissive Setpoint lower. (wrong direction) 2. The SBCS system can still generate a Auto Permissive Signal when real Steam Pressure goes above setpoint.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Main Turbine Generator: Knowledge of MT/G system design feature(s) and/or interlock(s) which provide for the following: Programmed controller for relationship between steam pressure at T/G inlet (impulse, first stage) and plant power level	Tier	2		
	Group	2		
	K/A	045 K4.01		
	IR	2.7		

Question 62

Given the following conditions:

- Unit 2 is operating at 100% power
- All inputs at the RRS Cabinet are selected to AVERAGE
- CEDMCS Mode Selector Switch is in AUTO

Subsequently:

- ONE of the two Turbine First Stage Pressure inputs to the RRS system begins to fail LOW resulting in Tref lowering at a rate of 1°F/min

Over the next 10 minutes, with NO operator action, how will the RRS system respond to this failure?

A. CEAs will remain ARO due to an AMI signal being generated prior to any CEA movement

B. CEAs will start inserting when Tavg-Tref difference reaches 3°F and continue inserting until operator action is taken

C. CEAs will start inserting when Tavg-Tref difference reaches 3°F, then stop inserting when the Tavg-Tref HI-LO alarm annunciates

D. CEAs will remain ARO due to the failed First Stage Pressure instrument being “kicked out” of the comparison circuit prior to any CEA movement

Proposed Answer:	A

Explanations:	
A.	Correct. AMI is generated when TLI signals (which are linearly proportional to TFSP signals) deviate by 5% and by the time CEA auto demand occurs, TLI signals would be deviating by ~ 13%.
B.	Plausible since this would occur if both TLIs were failing low, however since only one TFST pressure is failing low, CEAs will not insert.
C.	Plausible since CEAs would normally insert if TLI is failing low, however only if both TLIs were failing, and the Tavg-Tref HI-LO alarm generates an AWP which stops CEA movement, but only in the outward direction.
D.	Plausible since this is the case in other systems, such as DFWCS, however the RRS does not have an “auto kickout” for failing inputs such as TFSP.

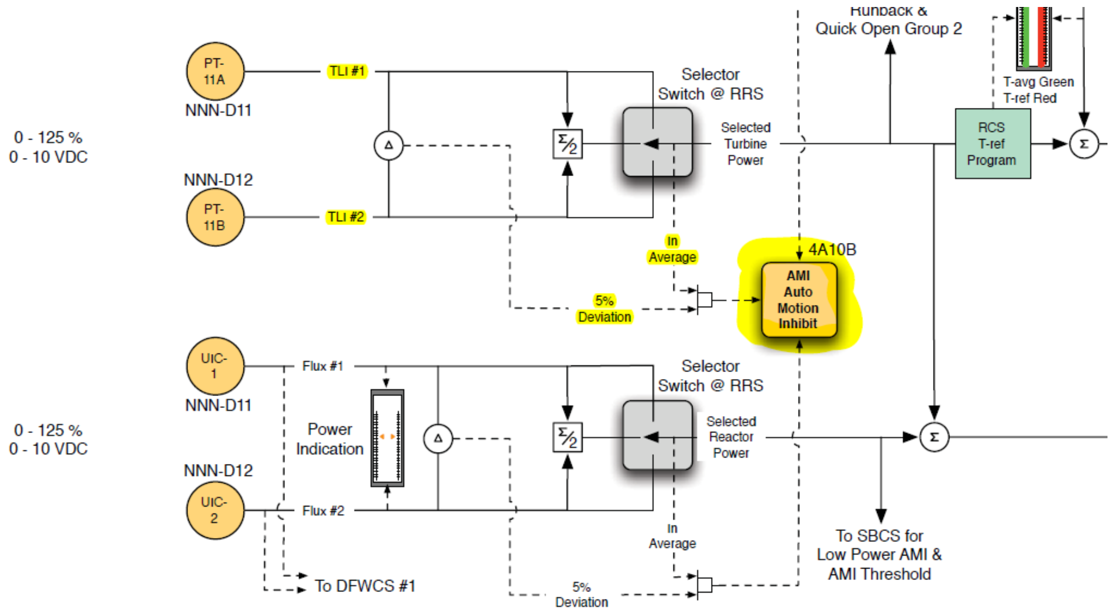
Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	4	
10CFR55.41:	7	
Reference Provided:	N	
Learning Objective:	Describe the response of the Reactor Regulating System to a failure of a Turbine Load Index (TLI) input.	

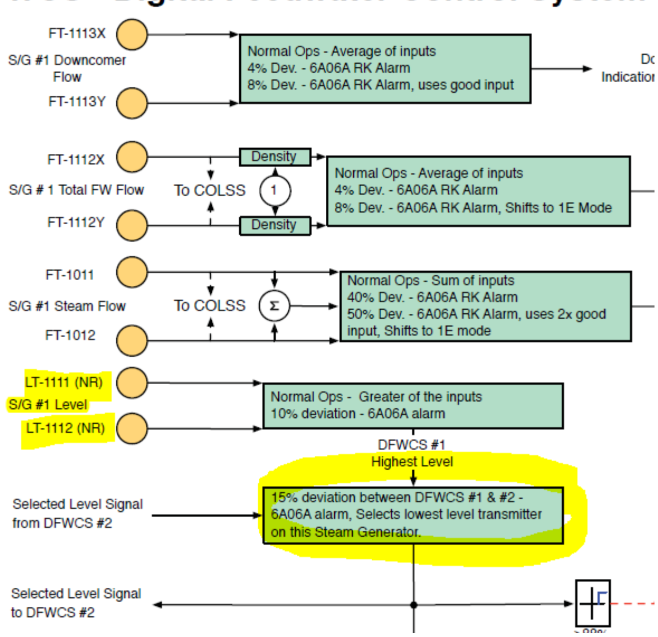
Reactor Regulating System uses two Turbine Load Index inputs (TLI #1 and #2) for indication of Turbine First Stage Pressure. This is then converted to Tref. If one TLI fails low, while selected to average, and RRS senses a 5% difference between TLI #1 and #2, an AMI signal will be generated. The AMI signal will block automatic CEA motion.

If both TLI inputs failed low OR if the failing TLI input was selected (not in average), RRS would sense the lowering TLI / Tref and cause an automatic CEA insertion.



Other control systems (i.e. DFWCS SG level input) have the ability to detect a deviation between two inputs and automatically select another instrument to use. DFWCS would continue to operate in Automatic if one input of SG level failed.

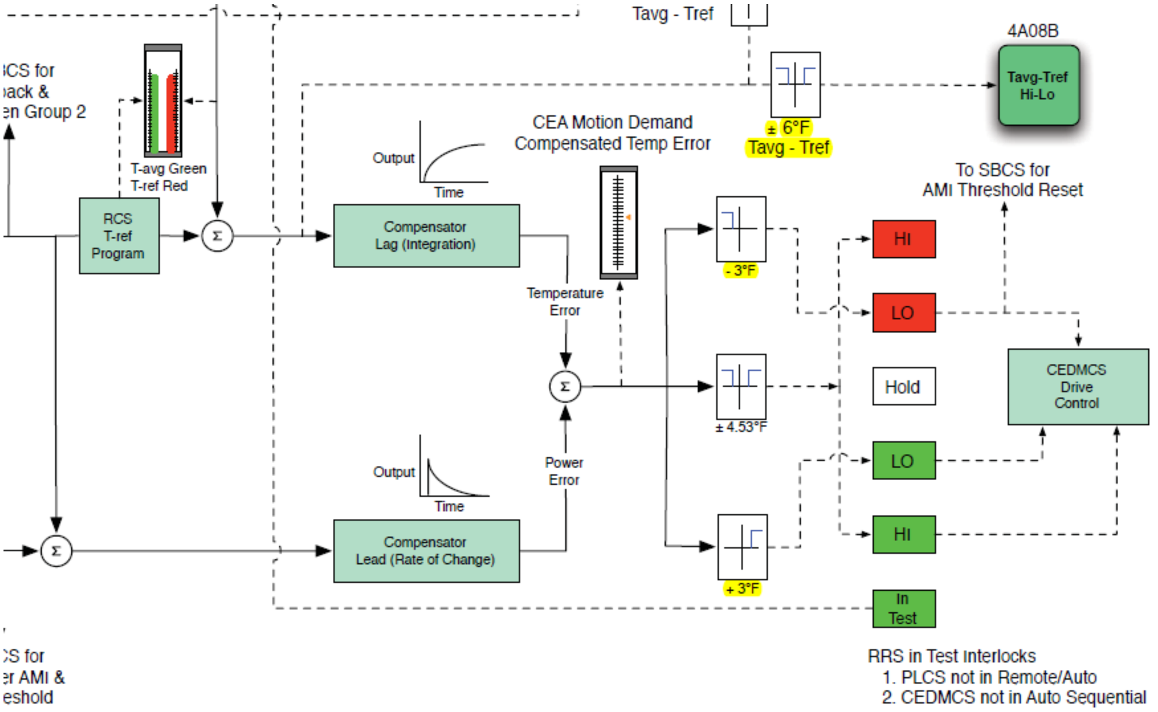
DFWCS - Digital Feedwater Control System



CEA motion demand first occurs at a ± 3°F temperature deviation.

Tavg-Tref Hi-Lo alarm comes in at ± 6°F temperature deviation.

The 5% deviation between TLI inputs will be sensed prior to reaching either of these temperature deviations, therefore an AMI will be generated and CEA motion will not occur.



Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Condenser Air Removal: Ability to perform specific system and integrated plant procedures during all modes of plant operation	Tier	2	[redacted]	
	Group	2	[redacted]	
	K/A	055 G 2.1.23		
	IR	4.3	[redacted]	

Question 63

Given the following conditions:

- Unit 2 is operating at 100% power
- Main Condenser vacuum is degrading
- Current Condenser vacuum:
    - ‘A’ Shell: 4.4 in HgA
    - ‘B’ Shell: 4.6 in HgA
    - ‘C’ Shell: 4.8 in HgA

Based on these conditions, with NO operator action, the ‘D’ Air Removal Pump should be running…

A. but not yet aligned to any Condenser shells

B. and the suction should be aligned to the ‘A’ Condenser shell ONLY

C. and the suction should be aligned to the ‘C’ Condenser shell ONLY

D. and the suction should be aligned to all three Condenser shells

Proposed Answer:	B

Explanations:	
A.	Plausible if thought that the ‘D’ AR Pump must be manually aligned to the shell(s) from which to take a suction, however given the listed shell pressures, the pump will automatically align to the ‘A’ shell
B.	Correct.
C.	Plausible since the ‘C’ shell is the lowest vacuum, however the ‘D’ AR Pump will not take a suction on the ‘C’ shell until pressure degrades to 5.3” HgA.
D.	Plausible since all 3 shells show degraded vacuum, and the ‘D’ pump will eventually align to all 3 shells, however at the listed shell pressures, the pump will only be aligned to the ‘A’ shell.

Question Source:		New	
	x	Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	x	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	4	
Reference Provided:	N	
Learning Objective:	Describe what components are impacted as backpressure rises.	

Response Section

7A01A
AIR REM SYS TRBL

Condenser Air Removal System Trouble

Point ID	Description	Setpoint
ARYS14	Condenser A Suction Valve Auto Open	NA
ARYS15	Condenser B Suction Valve Auto Open	NA
ARYS16	Condenser C Suction Valve Auto Open	NA

AUTOMATIC ACTION

- Air Removal pump ARN-P01D starts and the suction valve to the condenser with low vacuum opens.

MANUAL ACTIONS

___ 1. Ensure BOTH of the following:

____ • ARN-P01D, D Air Removal Vacuum Pump, is running

____ • The condenser suction valve in alarm is open

___ 2. Check individual condenser pressure indications:

Condenser Pressure Indicator ID	Suction Valve Auto Open Setpoint	Location
CDN-PI-47 - Condenser A Vacuum	4.2 in H_gA	B07
CDN-PI-48 - Condenser B Vacuum	4.8 in H_gA	B07
CDN-PI-49 - Condenser C Vacuum	5.3 in H_gA	B07

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Circulating Water: Knowledge of bus power supplies to the following: Emergency/essential SWS pumps	Tier	2		
	Group	2		
	K/A	075 K2.03		
	IR	2.6		

Question 64

(1) The feeder breaker for the 'A' Spray Pond Cooling Pump, SPA-P01, is located on…

(2) The feeder breaker for the 'A' Plant Cooling Water Pump, PWN-P01A, is located on…

A. (1) PBA-S03
(2) NAN-S01

B. (1) PBA-S03
(2) NBN-S01

C. (1) PGA-L31
(2) NAN-S01

D. (1) PGA-L31
(2) NBN-S01

Proposed Answer:	**B**

Explanations:	
A.	First part is correct. Second part is plausible as the Circulating Water Pumps are powered from NAN buses, however the Plant Cooling Water Pumps are powered from NBN.
B.	Correct.
C.	First part is plausible since there are class pumps powered from PGA-L31 such as the ‘A’ Charging Pump and the ‘A’ Fuel Pool Cooling Pump, however the ‘A’ Spray Pond Pump is powered from PBA-S03. Second part is plausible as the Circulating Water Pumps are powered from NAN buses, however the Plant Cooling Water Pumps are powered from NBN.
D.	First part is plausible since there are class pumps powered from PGA-L31 such as the ‘A’ Charging Pump and the ‘A’ Fuel Pool Cooling Pump, however the ‘A’ Spray Pond Pump is powered from PBA-S03. Second part is correct.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	**X**	**Memory or Fundamental Knowledge**
		Comprehension or Analysis

Level of Difficultly:	**2**	
10CFR55.41:	**7**	
Reference Provided:	**N**	
Learning Objective:	**Describe how the Class IE Electrical Distribution System supports the operation of the following systems: Essential Spray Pond System**	

Technical Reference:	40OP-9SP01 Essential Spray Pond (SP) Train A

Appendix A - Spray Pond A Electrical Lineup

Number	Name	Location	Electrical Drawing	Position	First Verif	Sec Verif
PBA-S03C	Essential Spray Pond Pump, SPA-P01	Control Building 100 ft PBA-S03	PBA-001	Racked In		

Technical Reference:	40OP-9PW01 Plant Cooling Water

Number	Name	Location	Drawing	Required Pos	Verified By Initial
NBN-S01H	4.16 KV SWGR Cubicle for PWN-P01A	NBN-S01 in Switchgear Room	PWB-001	Racked In & Open	

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Nuclear Instrumentation: Ability to manually operate and/or monitor in the control room: Trip Bypasses	Tier	2		
	Group	2		
	K/A	015 A4.03		
	IR	3.5		

Question 65

During a Reactor startup, the High Log Power Trips ___(1)___ bypassed as soon as power reaches ___(2)___ .

A. (1) are automatically
   (2) 1×10^{-2}

B. (1) are automatically
   (2) 1×10^{-4}

C. (1) may be manually
   (2) 1×10^{-2}

D. (1) may be manually
   (2) 1×10^{-4}

Proposed Answer:	D

Explanations:	
A.	First part is plausible since the bypass is automatically removed during a shutdown, however it must be manually bypassed on a startup. Second part is plausible since 1x10-2 is the trip setpoint for the high log power trip.
B.	First part is plausible since the bypass is automatically removed during a shutdown, however it must be manually bypassed on a startup. Second part is correct.
C.	First part is correct. Second part is plausible since 1x10-2 is the trip setpoint for the high log power trip.
D.	Correct.

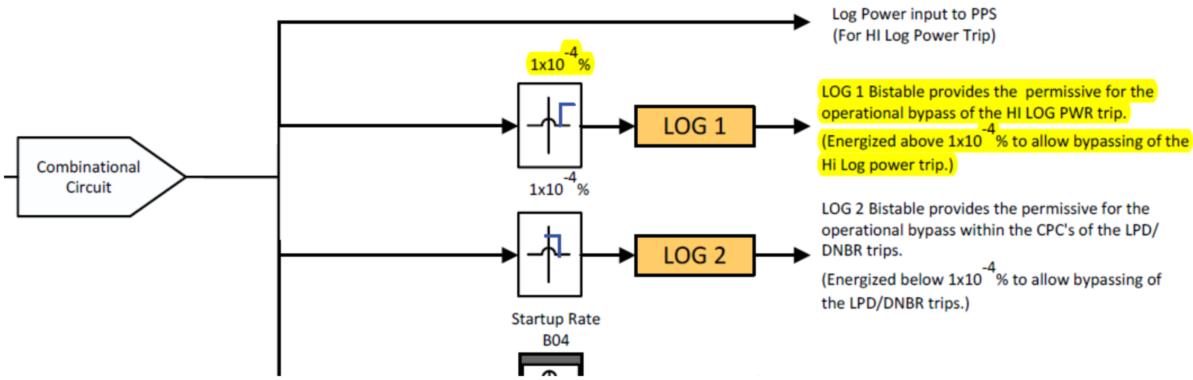
Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	Describe the approach to the POAH, including what actions are performed once this power level is reached.	

Technical Reference:	PVGS Operator Information Manual

Log 1 bistable energized as power rises above 10E-4%. Once energized, it allows the Hi Log Power reactor trip to be bypassed.



Technical Reference:	40OP-9ZZ03 Reactor Startup

Manual bypass of the hi log power trip is performed once the permissive alarm comes in.

6.5 Post Criticality Actions

6.5.11 **WHEN** the High Log Power Trip Bypass permissives are received,
THEN perform Appendix F - Bypassing High Log Power Trips.

Appendix F - Bypassing High Log Power Trips

NOTE

___ Alarm Window HI LOG PWR LVL BYP PERM (5A15B), is expected to alarm when the High Log Power Bypass Permissive light illuminates.

1.0 **WHEN** the High Log Power Bypass Permissive light illuminates,
THEN bypass the High Log Power Trips for each Log Safety Channel on B05:

Log Safety Channel	High Log Power Trip Position	Positioned By Initials	Ind. Verified By Initials
A	Bypassed		
B	Bypassed		
C	Bypassed		
D	Bypassed		

End of Appendix F

Response Section

5A15B
HI LOG PWR LVL BYP PERM

High Log Power Level Bypass Permissive

Point ID	Description	Setpoint
SBJS10A	Hi Log Power Level Bypass Permissive Channel A	10^{-4} Percent Power
SBJS10B	Hi Log Power Level Bypass Permissive Channel B	10^{-4} Percent Power
SBJS10C	Hi Log Power Level Bypass Permissive Channel C	10^{-4} Percent Power
SBJS10D	Hi Log Power Level Bypass Permissive Channel D	10^{-4} Percent Power

AUTOMATIC ACTION

NOTE

The Hi Log Power trip and pre-trip bypasses are automatically removed when logarithmic power level decreases below $l0^{-4}$ percent neutron rated thermal power.

- Actuates circuitry to allow manual bypassing of the Hi Log Power trips and pre-trips when reactor power level is greater than $l0^{-4}$ percent neutron rated thermal power.

Response Section

5A15C
HI LOG PWR LVL CH TRIP

High Log Power Level Channel Trip

Point ID	Description	Setpoint
SBTA02	Hi Log Power Level Channel A Trip	10^{-2} variable
SBTB02	Hi Log Power Level Channel B Trip	10^{-2} variable
SBTC02	Hi Log Power Level Channel C Trip	10^{-2} variable
SBTD02	Hi Log Power Level Channel D Trip	10^{-2} Variable

AUTOMATIC ACTION

- Reactor Trip on two or more channels

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Knowledge of the purpose and function of major system components and controls	Tier	3		
	Group			
	K/A	G 2.1.28		
	IR	4.1		

Question 66

Prior to restoring power to a Class 4.16 kV Bus from the opposite Train Emergency Diesel Generator, all pumps on the bus to be energized should have ___(1)___ in order to prevent ___(2)___ .

A. (1) their handswitches placed in Pull-to-Lock
   (2) overloading the EDG

B. (1) their handswitches place in Pull-to-Lock
   (2) an uncontrolled restoration of flow to each associated system

C. (1) an 86 lockout relay actuated on the pump breaker
   (2) overloading the EDG

D. (1) an 86 lockout relay actuated on the pump breaker
   (2) an uncontrolled restoration of flow to each associated system

Proposed Answer:	C

Explanations:	
A.	First part is plausible as this is done on some loads, such as ventilation ACUs, however the pumps have an 86 lockout actuated at the breaker. Second part is correct.
B.	First part is plausible as this is done on some loads, such as ventilation ACUs, however the pumps have an 86 lockout actuated at the breaker. Second part is plausible as this is done in some instances, i.e. to prevent an uncontrolled restoration of Seal Injection when power is restored to Charging Pumps, however the basis for this situation is to prevent overloading an EDG.
C.	Correct.
D.	First part is correct. Second part is plausible as this is done in some instances, i.e. to prevent an uncontrolled restoration of Seal Injection when power is restored to Charging Pumps, however the basis for this situation is to prevent overloading an EDG.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	7	
Reference Provided:	N	
Learning Objective:	From memory, describe the use of cautions and notes in the EOPs in accordance with 40DP-9AP15.	

Technical Reference:	40EP-9EO10-059 Appendix 59: Cross-tie DG A to PBB-S04

Prior to energizing the bus from an opposite train DG, the large loads on the bus will be disabled from auto starting by actuating their associated 86 lockout relay.

INSTRUCTIONS — CONTINGENCY ACTIONS

1. Direct an operator to PERFORM Attachment 59-A, Disable PBB-S04 Breakers.

Attachment 59-A — Disable PBB-S04 Breakers

INSTRUCTIONS — CONTING

____ 9. Manually trip the 86 relay for **ALL** of the following breakers on bus PBB-S04 (REFER TO Attachment 59-B, Manual Trip of 86 Relay):

- PBB-S04C, "ESSENTIAL SPRAY POND PUMP M-SPB-P01"
- PBB-S04D, "CONTAINMENT SPRAY PUMP M-SIB-P03"
- PBB-S04E, "H.P. SAFETY INJECT PUMP M-SIB-P02"
- PBB-S04F, "L.P. SAFETY INJECT PUMP M-SIB-P01"
- PBB-S04G, "ESSENTIAL CHILLER M-ECB-E01"
- PBB-S04M, "ESSENTIAL COOL WATER PUMP M-EWB-P01"
- PBB-S04S, "AUX FEED WATER PUMP M-AFB-P01"

INSTRUCTIONS	CONTINGENCY ACTIONS
6. **WHEN** informed by the area operator that the PBB-S04 breakers are disabled, **THEN** perform the following to close PBB-S04L from the Control Room:	____ 6.1 **IF** DC control power is **NOT** available to PBB-S04, **THEN** perform the following to energize PBB-S04, PKB-M42 and PNB-D26:
a. Place synchronizing switch PBB-SS-S04L, 4.16 KV Bus S04 Alternate Supply, to "ON".	a. **IF** the local closing spring indicator for PBB-S04L does **NOT** indicate "CHGD", **THEN** locally perform the following:
b. Close breaker PBB-S04L, 4.16 kV Bus S04 Alternate Supply.	1) Obtain a ratchet, extension and 5/8 inch socket from FPN-C02, "EMERGENCY
c. Place synchronizing switch PBB-SS-S04L to "OFF".	

Smaller 480V loads that may auto start are placed in PTL with the MCR hand switch.

4. Place **ALL** of the following in "PULL TO LOCK":
   - Train B Containment Normal ACUs
   - Train B CEDM ACUs

Technical Reference:	40DP-9AP17 Standard Appendices Technical Guideline

The reason for tripping the 86 lockouts is to prevent overloading the DG, since it will be supplying two 4160V busses. Prior to starting loads, the operators will ensure that DG will not be overloaded.

4.1.59 Appendix 59 - Cross-Tie DG A to PBB-S04

A. This appendix is used during a loss of offsite power event when Diesel Generator B fails to start and load its 4.16 KV bus. The appendix will align the Train A DG to PBB-S04 for the purpose of energizing class battery chargers. Breaker(s) may need to be operated locally if control power is not available which could occur from a loss of offsite power during an outage when battery maintenance is taking place and one battery is disconnected from its bus. Additional loads may be started if needed and if they are within the capability of the DG. The appendix can also be used if the Maintenance of Vital Auxiliaries Safety Function is lost because Train B equipment needed to maintain the Safety Function is not energized.

Attachment 59-A directs the actions that will be taken by an operator in the field to prepare the Train B vital 4.16 KV bus to be energized from DG A. These actions include placing the B Diesel Generator in off, checking for tripped protective relays and manually tripping the 86 lockout relays for the major loads on PBB-S04. Tripping the lockout relays allows this equipment to be reset and started from the Control Room.

Technical Reference:	40EP-9EO07 Loss of Offsite Power / Loss of Forced Circulation EOP

If all charging is lost due to a LOOP, flow restoration must be controlled to prevent damaging the seals. However, this is not a reason for disabling PB loads prior to energizing a bus from an opposite train DG.

CAUTION

Starting a Charging Pump without first isolating seal injection will likely cause seal damage due to rapid cooldown of the seals.

* 5. **IF** a LOOP has occurred,
THEN perform the following:

a. **IF** no charging pumps are running,
THEN isolate seal injection.

b. **IF** seal injection is isolated,
THEN isolate controlled bleedoff.

c. Reset the anti pump condition on the always running Charging Pump by placing the handswitch in the "STOP" position.

c.1 Select a new always runnin
Charging Pump.

c.2 **IF** the new always running
Charging Pump was the no

Examination Outline Cross-Reference:	**Level**	**RO**		**SRO**
K/A: Knowledge of the station's requirements for verbal communications when implementing procedures	**Tier**	**3**		
	Group			
	K/A	**G 2.1.38**		
	IR	**3.7**		

Question 67

Per 40DP-9OP02, Conduct of Operations, the "**A**" in which of the following plant components is REQUIRED to be verbally communicated as "**ALPHA**"?

1. PB**A**-S03
2. **A**FB-P01
3. Train '**A**' EDG

A. 1 and 2 ONLY

B. 1 and 3 ONLY

C. 2 and 3 ONLY

D. 1, 2, AND 3

Proposed Answer:	B

Explanations: All three are plausible if the reason for the letter A is unknown, or if the reasons for using the phonetic pronunciation are unknown, however only 1 and 3 require the use of the phonetic alphabet when spoken.	
A.	See above.
B.	Correct.
C.	See above.
D.	See above.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	Describe operations expectations when it comes to Communications in accordance with 40DP-9OP02, Conduct of Operations.	

Technical Reference:	40DP-9OP02, Conduct of Operations

4.3.2.7 The phonetic alphabet is used when referring to component, channel or train references with the sole use of the letter designation (e.g., train alpha) or when component, channel or train are referenced with the last letter of the component ID (class electrical bus PB Alpha). The phonetic alphabet is used when part of the message contains a single letter of the alphabet (e.g., alpha, bravo). The phonetic alphabet is as follows:

A-Alpha	J-Juliet	S-Sierra
B - Bravo	K - Kilo	T - Tango
C -Charlie	L - Lima	U - Uniform
D - Delta	M - Mike	V - Victor
E - Echo	N - November	W - Whiskey
F - Foxtrot	O - Oscar	X - X-ray
G - Golf	P - Papa	Y - Yankee
H - Hotel	Q - Quebec	Z - Zulu
I - India	R - Romeo	

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels	**Tier**	**3**		
	Group			
	K/A	**G 2.2.2**		
	IR	**4.6**		

Question 68

During a unit startup, the FIRST Main Feedwater Pump is placed in service ___(1)___ MODE 1 is entered, and the Main Turbine is synchronized to the grid ___(2)___ DFCWS goes through swapover.

A. (1) after
   (2) after

B. (1) after
   (2) before

C. (1) before
   (2) after

D. (1) before
   (2) before

Proposed Answer:	D

Explanations:	
A.	First part is plausible since the first MFP is placed in service prior to placing the Main Turbine online, which occurs at 12% power, and MODE 1 is entered at 5% power, however the capacity of the Startup AFW Pump is only sufficient to ~3% power so the MFP must be placed in service prior to MODE 1 being entered. Second part is plausible since going through swapover is required in order to be in 3 element feed control and it would make sense to have more responsive feedwater control prior to placing the Main Turbine online, however the Main Turbine is synched to the grid at 12% power and swapover doesn't occur until ~15% power.
B.	First part is plausible since the first MFP is placed in service prior to placing the Main Turbine online, which occurs at 12% power, and MODE 1 is entered at 5% power, however the capacity of the Startup AFW Pump is only sufficient to ~3% power so the MFP must be placed in service prior to MODE 1 being entered. Second part is correct.
C.	First part is correct. Second part is plausible since going through swapover is required in order to be in 3 element feed control and it would make sense to have more responsive feedwater control prior to placing the Main Turbine online, however the Main Turbine is synched to the grid at 12% power and swapover doesn't occur until ~15% power.
D.	Correct.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	Given key parameter indications and various plant conditions, predict plant operations during a plant startup in accordance with 40OP-9ZZ04, Mode 2 to Mode 1.	

Technical Reference:	40OP-9ZZ04 Plant Startup Mode 2 To Mode 1

Overall sequence shown below (start 1st MFP in Mode 2 (< 5% RTP), place turbine / generator in-service at ~12%, then raise power through feedwater control swapover which occurs between 15-16.5%.)

6.20 Commence raising reactor power to 2%.

6.28 Start a Main Feedwater Pump (MFWP) per ONE of the following:

____ • 40OP-9FT01, Feedwater Pump Turbine A

____ • 40OP-9FT02, Feedwater Pump Turbine B

6.39 Raise reactor power to 5%.

CAUTION

___ Failure to maintain reactor power less than 14% as indicated on Control Channel NIs could result in premature feedwater swapover.

6.57 Commence raising reactor power to 11.5% to 12.5%.

6.61 Complete Main Turbine startup per 40OP-9MT02, Main Turbine.

6.70.4 Place the Main Generator in service per 40OP-9MB01, Main Generator and Excitation.

NOTE

___ Swapover will occur when the Control Channel NIs indicate reactor power is greater than 16.5% or between 15% and 16.5% with either downcomer feed regulating valve greater than 80% open.

6.83 Perform ONE of the following appendices to control the plant during feedwater swapover:

____ • Appendix A - Guidelines For Downcomer To Economizer Swapover In Auto

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Knowledge of tagging and clearance procedures	**Tier**	**3**		
	Group			
	K/A	**G 2.2.13**		
	IR	**4.1**		

Question 69

Per 40DP-9OP29, Power Block Clearance and Tagging, which of the following evolutions are performed using CONCURRENT verification?

1. Replacement of a danger tag which has become damaged
2. Moving a danger tag from a breaker hasp to the cubicle door
3. Initial hanging of a danger tag on a 480V breaker for corrective maintenance

A. 1 and 2 ONLY

B. 1 and 3 ONLY

C. 2 ONLY

D. 3 ONLY

Proposed Answer:	A

Explanations:	
A.	Correct.
B.	Plausible since replacing a tag is done using CV, and it is plausible that initial tag hanging would be done using CV to ensure it is done correctly the first time, and for operators who have been qualified less than one year, a peer check is required for initial tag hanging, however this is still done using IV.
C.	Plausible since moving a tag is done using CV, however replacement of a tag is also done using CV.
D.	Plausible that initial tag hanging would be done using CV to ensure it is done correctly the first time, and for operators who have been qualified less than one year, a peer check is required for initial tag hanging, however this is still done using IV.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	Given a need for personnel or equipment protection, the operator will determine the Clearance and Tagging requirements per Palo Verde procedures.	

Technical Reference:	40DP-9OP29 Power Block Clearance and Tagging

Concurrent Verification is used for replacing a damaged tag.

4.11.3 Replacing Tags

A. If a clearance tag is worn or illegible, then Operations shall perform the following:

1. Create a duplicate tag with the same clearance and tag number as the worn tag.

2. If the tag is a credited tag, then print the new tag from the original clearance from the WMPTHIST or WMPTMAIN screen.

3. Reattach the tag using concurrent verification.

Concurrent Verification is used for moving a tag to a different location on the same component.

4.11.5 Moving Tags

3. The tag removal and tag attachment is verified with concurrent verification.

Independent verification is used for establishing a clearance for maintenance.

4.7 Verifying a Clearance

4.7.2 The Independent Verifier shall check the tag(s) are hanging on the right equipment per 02DP-9OP01, Site Wide Status Control Procedure.

Technical Reference:	40DP-9OP02 Conduct of Operations

Peer checks may be required for operating breakers, however, the IV for the clearance is still required in addition to the peer check.

4.7.8 Breaker Manipulation Peer Check Practices

A. All the following breakers require peer checks during manipulations unless otherwise described below:

- Class Power Breakers
- Non Class 13.8KV and 4160V Breakers
- Non Class DC M45 and M46 Breakers
- Non Class 120 AC breakers in ANY of the following:
  - NNN-D11
  - NNN-D12
  - NNN-D15
  - NNN-D16

B. Items NOT identified in STEP 4.7.8.A are peer checked at the discretion of the SM/CRS/WCSRO. The determination is based on their knowledge and BOTH of the following criteria:

- Proficiency of the performers.
- Integrated risk to the unit.

Examination Outline Cross-Reference:	**Level**	**RO**	[redacted]	**SRO**
K/A: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations	**Tier**	**3**	[redacted]	
	Group		[redacted]	
	K/A	**G 2.2.36**		
	IR	**3.1**	[redacted]	

Question 70

Given the following conditions:

- Maintenance is about to perform preventive maintenance on a Technical Specification piece of equipment
- The equipment will be inoperable during the duration of the maintenance
- The configuration of the equipment during the maintenance will NOT automatically bring in any alarms on the SESS panel
- **At time = 1200:** The SM declared the associated equipment inoperable

Per 40DP-9OP02, Conduct of Operations, the crew is required to insert a manual SESS alarm NO LATER THAN ___(1)___ and when the manual SESS is inserted, the affected equipment will have ___(2)___ light illuminated on the SESS panel.

A. (1) 1800
   (2) ONLY a white

B. (1) 1800
   (2) a white AND blue

C. (1) 2400
   (2) ONLY a white

D. (1) 2400
   (2) a white AND blue

Proposed Answer:	A

Explanations:	
A.	Correct.
B.	First part is correct. Plausible that manual insertion of a SESS alarm would illuminate both halves of the component SESS alarm window, however when inserting a manual SESS alarm, only the white alarm is illuminated.
C.	First part is plausible since 2400 is 12 hours from the time of inoperability, which is the duration of a shift, however the requirement is to install a manual SESS alarm prior to the end of the current shift, which is 1800. Second part is correct.
D.	First part is plausible since 2400 is 12 hours from the time of inoperability, which is the duration of a shift, however the requirement is to install a manual SESS alarm prior to the end of the current shift, which is 1800. Second part is plausible that manual insertion of a SESS alarm would illuminate both halves of the component SESS alarm window, however when inserting a manual SESS alarm, only the white alarm is illuminated.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	**X**	**Memory or Fundamental Knowledge**
		Comprehension or Analysis

Level of Difficultly:	**3**	
10CFR55.41:	**10**	
Reference Provided:	**N**	
Learning Objective:	**Describe when a manual Safety Equipment Safety System (SESS) alarm is required to be inserted.**	

Technical Reference:	40DP-9OP02 Conduct of Operations

V. Manual Safety Equipment Status System (SESS) Inputs

1. A manual SESS alarm input shall be initiated when any ES annunciator panel monitored component or system is ONE of the following:

   - Disabled and unable to perform the intended function by any method that is NOT alarmed; or

   - Rendered incapable of performing the intended design function by any method that is NOT alarmed.

NOTE

The impairment may be the result of a Permit, T-mod., procedural alignment, or component failure.

2. The manual SESS alarm is required if the impairment or failure is anticipated to last longer than the current shift.

   a) The manual SESS alarm should be initiated at the time of the occurrence but in all cases shall be in place prior to shift turnover.

3. Document initiation and removal of all SESS manual inputs in the Unit Log stating why the input is required or removed.

4. Initiate a Technical Specification Component Condition Record (TSCCR) if applicable.

5. The CRS shall be informed of all SESS panel changes.

Technical Reference:	40OP-9SI02 Recovery From Shutdown Cooling To Normal Operating Lineup

Example: Filling a CS header makes CS inoperable, but an automatic SESS alarm would not be actuated. The procedure directs a manual SESS input.

6.26 Filling the Train A Containment Spray Header for Evaporation Losses or during the Monthly Venting from the Penetration Room

6.26.9 **IF** in MODE 1, 2, 3, or 4,
THEN notify SM/CRS that Containment Spray System Train A is about to be made inoperable per LCO 3.6.6, Containment Spray System, by installing a hose connection at SIE-V502, Isolation Valve for CS Header Fill Line A.

6.26.10 **IF** in MODE 1, 2, 3, or 4,
THEN insert a manual Containment Spray A Safety Equipment Status System (SESS) alarm.

Technical Reference:	40AL-9ES2A Safety Equipment Status System Panel ESA-UA-2A Alarm Responses

As an example, if a manual Train A Containment Spray SESS alarm was inserted, the white alarm shown below would be illuminated.

Alarm Index

Containment Spray

ES2A03B
CNTMT SPRAY

Point ID		Description	Page
SEAS	16J	CS PMP A P03	439
SEIS	16J	CS PMP A P03	455

With the manual SESS inserted, only the white alarm would be illuminated. The individual component indications would be dark. These would only be lit if the component failed to actuate when required (blue light) or if a failure is detected that would make the component inoperable (white light).

Response Section

Containment Spray Pump A P03

SEAS 16J
CS PMP A P03

PBA-S03D-752 contact	SIA-P03 Ckt Brk "A" Cntmt Spray Pump	SIA-P03 NOT running

Response Section

Containment Spray Pump A P03

SEIS 16J
CS PMP A P03

PBA-S03D-786	SIA-P03 Ckt Brk "A" Cntmt Spray Pump Lockout Relay	786 Relay tripped, SIA-P03 Inoperable
OR		
PBA-S03D-762T	Control Power and Breaker Racking Monitor	762T Relay de-energized SIA-P03 Inoperable
OR		
PBA-S03D-762C	Control Power, Breaker Racking Monitor, and Spring Charging Circuit Monitor	762C Relay de-energized SIA-P03 Inoperable

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Knowledge of radiation exposure limits under normal or emergency conditions	**Tier**	**3**		
	Group			
	K/A	**G 2.3.4**		
	IR	**3.2**		

Question 71

Per 10CFR20.1201, Occupational Dose Limits, the annual limit for TEDE for adults is ___(1)___ .

Per 75DP-9RP01, Radiation Exposure and Access Control, the INITIAL administrative exposure hold point for TEDE at PVNGS is ___(2)___ .

A. (1) 5 rem
   (2) 1.5 rem

B. (1) 5 rem
   (2) 2.5 rem

C. (1) 15 rem
   (2) 1.5 rem

D. (1) 15 rem
   (2) 2.5 rem

Proposed Answer:	A

Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible since 2.5 rem is the limit for does at both PV and other utilities combined, however an increase to that level requires additional management approvals.
C.	First part is plausible since 15 rem is a limit in 10CFR20, however that is the limit for exposure to the lens of the eye. Second part is correct.
D.	First part is plausible since 15 rem is a limit in 10CFR20, however that is the limit for exposure to the lens of the eye. Second part is plausible since 2.5 rem is the limit for does at both PV and other utilities combined, however an increase to that level requires additional management approvals.

Question Source:		New	
	X	Bank	
		Modified	
	x	Previous NRC Exam	2019 NRC Q71

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	12	
Reference Provided:	N	
Learning Objective:	State the plant administrative limits and guidelines for radiation dose	

Technical Reference:	75DP-9RP01 Radiation Exposure and Access Control

4.10 Radiation Exposure Limitations and Controls

4.10.1 10 CFR 20.1201 Occupational Dose Limits

A. Annual Occupational radiation dose to adults shall be limited to all of the following:

1. 5 rem total effective dose equivalent (TEDE) or 50 rem total organ dose equivalent (TODE), whichever is more limiting.

2. 15 rem lens dose equivalent (lens of the eye).

3. 50 rem shallow-dose equivalent (skin or any extremity).

Initial hold point is 1.5 rem/year TEDE.

4.10.4 Administrative Exposure Hold Points

A. To further maintain exposures ALARA, individuals are assigned an initial administrative exposure hold point of 1.5 rem/year TEDE.

1. To ensure exposures are kept ALARA, management must approve requests for assignment of higher administrative hold points.

Higher admin hold limits require additional approvals.

B. Approvals required per new limit request.

1. The Radiation Worker approval is required.

2. For a hold point up to 2000 mrem/year, RP Superintendent approval required.

3. For a hold point up to 2500 mrem/year, Radiation Protection Manager approval required.

4. For a hold point higher than 2500 mrem/year up to 4000 mrem/year, the ALARA Committee review and approval is required (as signified by the signature of an ALARA Committee Chairman).

5. For a higher administrative exposure hold point which would allow a worker's cumulative lifetime exposure (in rem) to exceed the worker's age (in years) are reviewed and approved by the ALARA Committee Chairman.

6. For any hold point that would cause an individual's exposure to exceed 10 rem cumulative site exposure within a 5 year period, approval of the Sr Vice President, Site Operations is required.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Ability to control radiation releases	**Tier**	**3**		
	Group			
	K/A	**G 2.3.11**		
	IR	**3.8**		

Question 72

In order to control magnitude of radioactive release during a SGTL, which of the following actions are directed per 40AO-9ZZ02, Excessive RCS Leakrate, Appendix C, Minimize Release to the Environment?

1. Ensure all available Cooling Tower Fans are in operation

2. Ensure closed then de-energize both Steam Supply Valves to AFA-P01

3. Place Steam Bypass Control Valves 1007 and 1008 Mode Select Switches to OFF

A. 1 and 2 ONLY

B. 1 and 3 ONLY

C. 2 ONLY

D. 3 ONLY

Proposed Answer:	**D**

Explanations:	
A.	Placing all cooling tower fans in operation is plausible as this would maximize vacuum in the Main Condenser and would aid in keeping any radioactivity inside the feed and condensate system, however maximizing vacuum would be done by placing all Air Removal Pumps in service. Closing and de-energizing the steam supply valves to AFA-P01 would aid in limiting release to the environment, however ensuring AFA-P01 is done by closing only the steam supply valve from the most affected SG.
B.	Placing all cooling tower fans in operation is plausible as this would maximize vacuum in the Main Condenser and would aid in keeping any radioactivity inside the feed and condensate system, however maximizing vacuum would be done by placing all Air Removal Pumps in service. Choice 3 is correct.
C.	Closing and de-energizing the steam supply valves to AFA-P01 would aid in limiting release to the environment, however ensuring AFA-P01 is done by closing only the steam supply valve from the most affected SG.
D.	Correct.

Question Source:	**x**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	**x**	**Memory or Fundamental Knowledge**
		Comprehension or Analysis

Level of Difficultly:	**3**	
10CFR55.41:	**12**	
Reference Provided:	**N**	
Learning Objective:	**Given indications of a Steam Generator Tube Leak, describe the possible adverse effects of SBCS and Aux. Steam operation, and the operator action to minimize these consequences.**	

Technical Reference:	40AO-9ZZ02 Excessive RCS Leakrate

Appendix C, Minimize Release to the Environment

INSTRUCTIONS	CONTINGENCY ACTIONS
3. **IF** Aux Feed Pump A is being used, **THEN** perform the following: a. Close the Aux Feed Pump A Steam Supply Valve from the affected Steam Generator. b. Notify Radiation Protection and the RMS Technician that Aux Feed Pump A is steaming to atmosphere.	
4. Select "OFF" on **BOTH** of the following switches: • SGN-HS-1007, Valve 7 Mode Select • SGN-HS-1008, Valve 8 Mode Select	

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	Tier	3	[redacted]	
	Group		[redacted]	
	K/A	G 2.3.15		
	IR	2.9	[redacted]	

Question 73

Given the following condition:

- An operator needs to exit the Aux Building to the yard to perform a local level check on the Refueling Water Tank

Prior to exiting the RCA, the operator must use a ___(1)___ to ensure they are free of contamination, and must contact RP if the reading on the device used in Part 1 exceeds a MINIMUM of ___(2)___ above background radiation.

(1) Hand-Held Frisker
(2) 100 cpm

(1) Hand-Held Frisker
(2) 200 cpm

(1) Personnel Contamination Monitor
(2) 100 cpm

(1) Personnel Contamination Monitor
(2) 200 cpm

Proposed Answer:	A

Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible 200 cpm is the limit for background radiation to use a frisker, however 100 cpm above background requires a call to RP.
C.	First part is plausible since the operator is leaving the Aux Building to go outside, and passing through a PCM is required when leaving the RCA, however when going from the Aux Building to the yard, only a hand held frisker is required to be used. Second part is correct.
D.	First part is plausible since the operator is leaving the Aux Building to go outside, and passing through a PCM is required when leaving the RCA, however when going from the Aux Building to the yard, only a hand held frisker is required to be used. Second part is plausible 200 cpm is the limit for background radiation to use a frisker, however 100 cpm above background requires a call to RP.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	x	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	12	
Reference Provided:	N	
Learning Objective:	Explain how to monitor personnel and personal items for contamination, including the use of friskers and personnel contamination monitors.	

Technical Reference:	

Although going to the RWT requires leaving the Aux Building, the RWT is still in an RCA and therefore a PCM is not used for leaving the building. This would only be required if the RWT was not in an RCA, or if there was a non-posted area between the aux building and the RWT, however it is continuously posted as an RCA

Contamination:

Using a Personnel Contamination Monitor

When preparing to leave the radiologically controlled area (RCA), you will be required to use a second type of contamination monitor, the personnel contamination monitor (PCM).

This monitor has several radiation detectors that monitor your entire body. It is more sensitive to radiation than a frisker and will detect very low levels of contamination.

That coupled with very large detectors makes the PCM the best way to find contamination on your body. Friskers are good at pin pointing where the contamination is located but PCMs are the best at finding out if contamination is present. They should be the very last instrument used before leaving.

Don't take personal items or sharp objects into the PCM, as they can puncture the detectors. Personal objects must be surveyed and released at the direction of RP personnel.

If the PCM alarms, exit in the same direction from which you entered. Do not exit into the clean area. Notify RP for assistance.

Personnel using one of several types of PCM's. Personnel contamination monitors are the best at detecting contamination. They have very large detectors and are very sensitive.

Technical Reference:	Radworker Training Lesson Plan

Although the limit for background radiation is 200 cpm, the minimum reading above background that requires a call to RP is 100 cpm.

Contamination:

Using a Frisker

You will use two main types of contamination monitors at any plant: a frisker and a personnel contamination monitor (PCM).

Friskers are the first line of defense.They are very easily set up anywhere frisking is needed. RP personnel place these in the plant in strategic locations to make sure contamination is not spread through the plant.

When exiting, personnel will use an automated personnel contamination monitor. Obey the signs and use the equipment provided as you walk around the plant.

Friskers use a hand-held probe coupled with a meter that you use to check yourself for contamination. Before you pick up the probe, check to ensure the frisker is turned on, set to the X1 scale and the display is less than 200 counts per minute (cpm). Monitor your hands by passing them, one at a time, about one half inch above the frisker probe at a speed of about 2 inches per second. Monitor the front and back of each hand.

Check the count rate on the meter. If it increases by more than 100 cpm and stays above that level, or if the monitor alarms, stay in the area and contact RP.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures	**Tier**	**3**		
	Group			
	K/A	**G 2.4.4**		
	IR	**4.5**		

Question 74

While operating at 100% power, which of the following equipment failures would warrant immediate entry into an Abnormal Operating Procedure (as opposed to an Alarm Response Procedure)?

A. Control Channel NI #1 fails to 50%

B. SG #2 Feed Flow transmitter, SGN-FT-1122, fails off-scale low

C. Letdown HX Outlet Temp Control, CHN-TIC-223, output fails to 100%

D. Containment NR Pressure transmitter, HCA-PI-351A, fails off-scale high

Proposed Answer:	A

Explanations:	
A.	Correct. This meets the entry conditions for 40AO-9ZZ16, RRS Malfunctions
B.	Plausible since there are AOPs for feed malfunctions such as 40AO-9ZZ09, RPCB (Loss of Feedpump), however failures that impact DFWCS are mitigated using the ARP.
C.	Plausible since failures of LDHX temp can result in a loss of letdown which would warrant entry into an AOP, however with output failing high, this would be addressed using the ARP.
D.	Plausible since this is an input to PPS and would warrant a TS call, however this failure on it's own would be addressed using the ARP.

Question Source:	x	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	x	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	Describe operations expectations when it comes to Procedure Use in accordance with 40DP-9OP02, Conduct of Operations.	

Technical Reference:	40AO-9ZZ16 RRS Malfunctions

ENTRY CONDITIONS

The RRS Malfunctions procedure may be entered when ANY of the following conditions exist:

3. Section 5.0, Control Channel NI Failures

- "AMI (AUTOMATIC MOTION INHIBIT)" annunciator (4A10B)
- "SU AND CONT CH TRBL" annunciator (4A12A)
- Channel Deviation Alarm on RRS Test Panel
- Comparison of indications on SEN-JR-5 or SEN-JI-7 indicates that one channel is failed

Technical Reference:	40AL-9RK6A Panel B06A Alarm Responses

Failed Feed Flow transmitter will be addressed using the ARP

Response Section

6A06A
FWCS PROCESS TRBL

Feedwater Control System Process Trouble

Point ID	Description	Setpoint
(x)FWCS1:B12	SG 1 Total Feedwater Flow 8% Deviation	8%
(x)FWCS2:B12	SG 2 Total Feedwater Flow 8% Deviation	8%
(x)FWCS1:B1D	SG 1 Steam Flow 50% Deviation	50%
(x)FWCS2:B1D	SG 2 Steam Flow 50% Deviation	50%

5. **IF** ANY of the following are at fault:

____ • FT1112X, Transmitter for Total FW Flow to S/G 1

____ • FT1112Y, Transmitter for Total FW Flow to S/G 1

____ • FT1122X, Transmitter for Total FW Flow to S/G 2

____ • FT1122Y, Transmitter for Total FW Flow to S/G 2

____ • FT1011, S/G 1 Line 1 Flow Transmttr

____ • FT1012, S/G 1 Line 2 Flow Transmttr

____ • FT1021, S/G 2 Line 1 Flow Transmttr

____ • FT1022, S/G 2 Line 2 Flow Transmttr

THEN perform the following at the DFWCS:

____ 5.1 Place the faulty transmitter in Maintenance.

____ 5.2 Adjust the affected Steam Generator level setpoint to match the actual affected Steam Generator level.

____ 5.3 Remove the DFWCS Three Element lockout.

____ 5.4 Reset the DFWCS alarm at the Process Alarm page.

Technical Reference:	40AL-9RK5A Panel B05A Alarm Responses

A failed CTMT pressure transmitter will be addressed by the ARP. It will direct placing the failed instrument in bypass to comply with Tech Specs. AOP entry is not required or directed.

Response Section

5A06C

HI CNTMT PRESS CH TRIP

High Containment Pressure Channel Trip

Point ID	Description	Setpoint
SBTA13	Hi Containment Pressure Ch A Trip	3 psig
SBTB13	Hi Containment Pressure Ch B Trip	3 psig
SBTC13	Hi Containment Pressure Ch C Trip	3 psig
SBTD13	Hi Containment Pressure Ch D Trip	3 psig

3. Compare ALL of the following Containment Pressure indications to confirm the alarm: (B05)

___ • HCA-PI-351A, Containment Pressure

___ • HCB-PI-351B, Containment Pressure

___ • HCC-PI-351C, Containment Pressure

___ • HCD-PI-351D, Containment Pressure

4. **IF** the alarm is confirmed to be valid (pressure rising),
THEN perform the following:

___ 4.1 Trip the Reactor

___ 4.2 GO TO 40EP-9EO01, Standard Post Trip Actions.

5. **IF** the alarm is confirmed to be invalid,
THEN place ANY affected channel in BYPASS at the associated Plant Protection System (PPS) cabinet:

___ • SBA-C01, Plant Protection Sys Cab

Technical Reference:	40DP-9OP02 Conduct of Operations

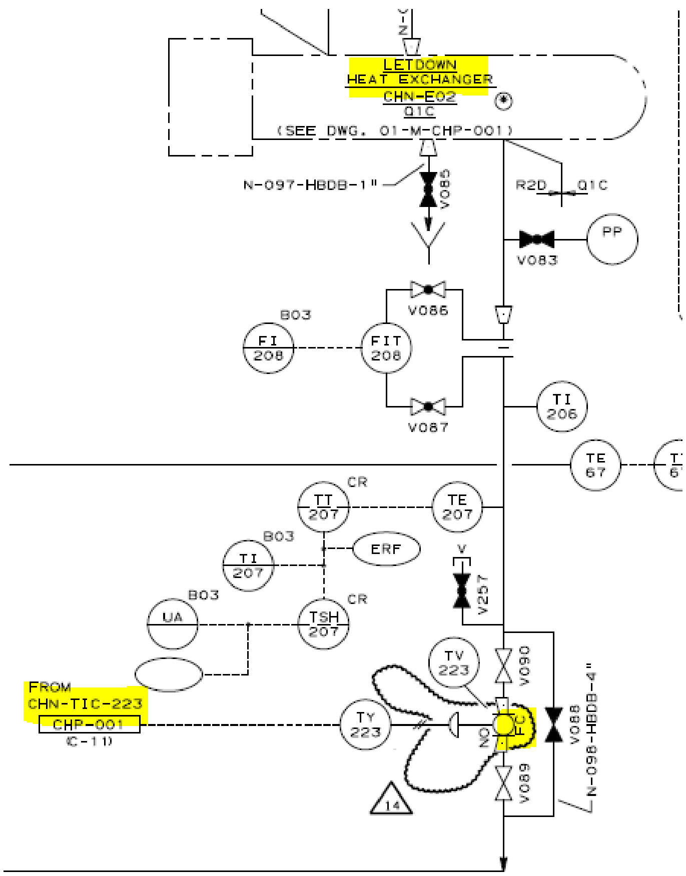
4.7.15 Manual Override of Automatic Systems

H. During equipment malfunctions which result in a transient, the Operator shall make a conscious decision on controller operation based on understanding of the transient and what manual actions are likely to be effective in mitigating the transient.

1. If the operator believes a transient can be controlled by taking manual control, then the operator should do so immediately and inform the other Control Room personnel of the action.
2. The CRS shall state acceptance or direct alternate actions.

Technical Reference:	Drawing 01-M-NCP-002 (NCW System P&ID)

NCN-TV-223 controls NCW flow through the Letdown HX. The TCV is a fail closed valve. With 100% output on the controller, it would be full open, and would cause letdown temp to drop. This could potentially result in a positive reactivity addition due to the letdown demins adsorbing more boron. Conduct of Ops allows operators to take manual control of the valve if auto has failed.



Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Knowledge of the emergency plan	Tier	3	[redacted]	
	Group		[redacted]	
	K/A	G 2.4.29		
	IR	3.1	[redacted]	

Question 75

Per EP-0905, Protective Actions:

(1) The LOWEST EAL level at which Accountability is REQUIRED inside the Protected Area is a(n) _____ .

(2) Accountability within the Protected Area must be completed within a MAXIMUM of _____ minutes of declaration of the EAL level listed in Part 1.

A. (1) Alert
(2) 15

B. (1) Alert
(2) 30

C. (1) Site Area Emergency
(2) 15

D. (1) Site Area Emergency
(2) 30

Proposed Answer:	D

Explanations:	
A.	First part is plausible since the Alert level is the lowest EAL level at which the ERO is activatied, however accountability is not required until the SAE level. Second part is plausible since 15 minutes is the time limit to make notifications to state and local agencies, however accountability is not required for 30 minutes.
B.	First part is plausible since the Alert level is the lowest EAL level at which the ERO is activatied, however accountability is not required until the SAE level. Second part is correct.
C.	First part is correct. Second part is plausible since 15 minutes is the time limit to make notifications to state and local agencies, however accountability is not required for 30 minutes.
D.	Correct.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	Identify the requirements for initiating Assembly and Accountability	

Technical Reference:	EP-0905, Protective Actions

- Other similar events as deemed appropriate by the Emergency Coordinator.

3.1.10 Assembly is mandatory at the Site Area Emergency (SAE) or higher level classification. Assembly of site personnel outside the Protected Area is accomplished by all personnel reporting to designated assembly areas. Assembly may be initiated at any time site management deems it appropriate for personnel safety reasons. Assembly may also be used as a tool to initiate the Two-Man Rule during Security events.

3.1.11 Accountability within the Protected Area is mandatory at the Site Area Emergency. Accountability may be initiated at other times at the discretion of the Emergency Coordinator to support worker safety. Accountability of personnel within the Protected Area is accomplished within 30 minutes of the declaration of Site Area Emergency or higher, and maintained continuously thereafter, using Protected Area(s) boundary access control as described in the PVNGS Security Plan.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Small Break LOCA: Knowledge of the purpose and function of major system components and controls	**Tier**			**1**
	Group			**1**
	K/A	**009 G 2.1.28**		
	IR			**4.1**

Question 76

Given the following conditions:

- Unit 1 is operating at 100% power
- The Train ‘A’ HPSI Pump and the Train ‘B’ LPSI Pump are inoperable
- Both pumps are tagged out for emergent corrective maintenance

Subsequently:

- The Reactor was manually tripped due to a small break LOCA
- SIAS and CIAS were manually actuated

Based on the listed conditions, this accident ___(1)___ within the analyzed conditions in the PVNGS UFSAR, and following SPTAs, the CRS should transition to ___(2)___ .

A. (1) IS
(2) 40EP-9EO03, LOCA

B. (1) IS
(2) 40EP-9EO09, Functional Recovery

C. (1) is NOT
(2) 40EP-9EO03, LOCA

D. (1) is NOT
(2) 40EP-9EO09, Functional Recovery

Proposed Answer:	**A**

Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible given that neither Train A nor Train B has a complete train of SI available, however since there is one full train of flow available, the correct action would be to mitigate the event using the LOCA EOP.
C.	First part is plausible since the FSAR analyzes for events with corresponding equipment failures (i.e. loss of offsite power), and in this case BOTH trains of SI are degraded, however with one effective full train of SI available, this would still be considered an analyzed event. Second part is correct.
D.	First part is plausible since the FSAR analyzes for events with corresponding equipment failures (i.e. loss of offsite power), and in this case BOTH trains of SI are degraded, however with one effective full train of SI available, this would still be considered an analyzed event. Second part is plausible given that neither Train A nor Train B has a complete train of SI available, however since there is one full train of flow available, the correct action would be to mitigate the event using the LOCA EOP.

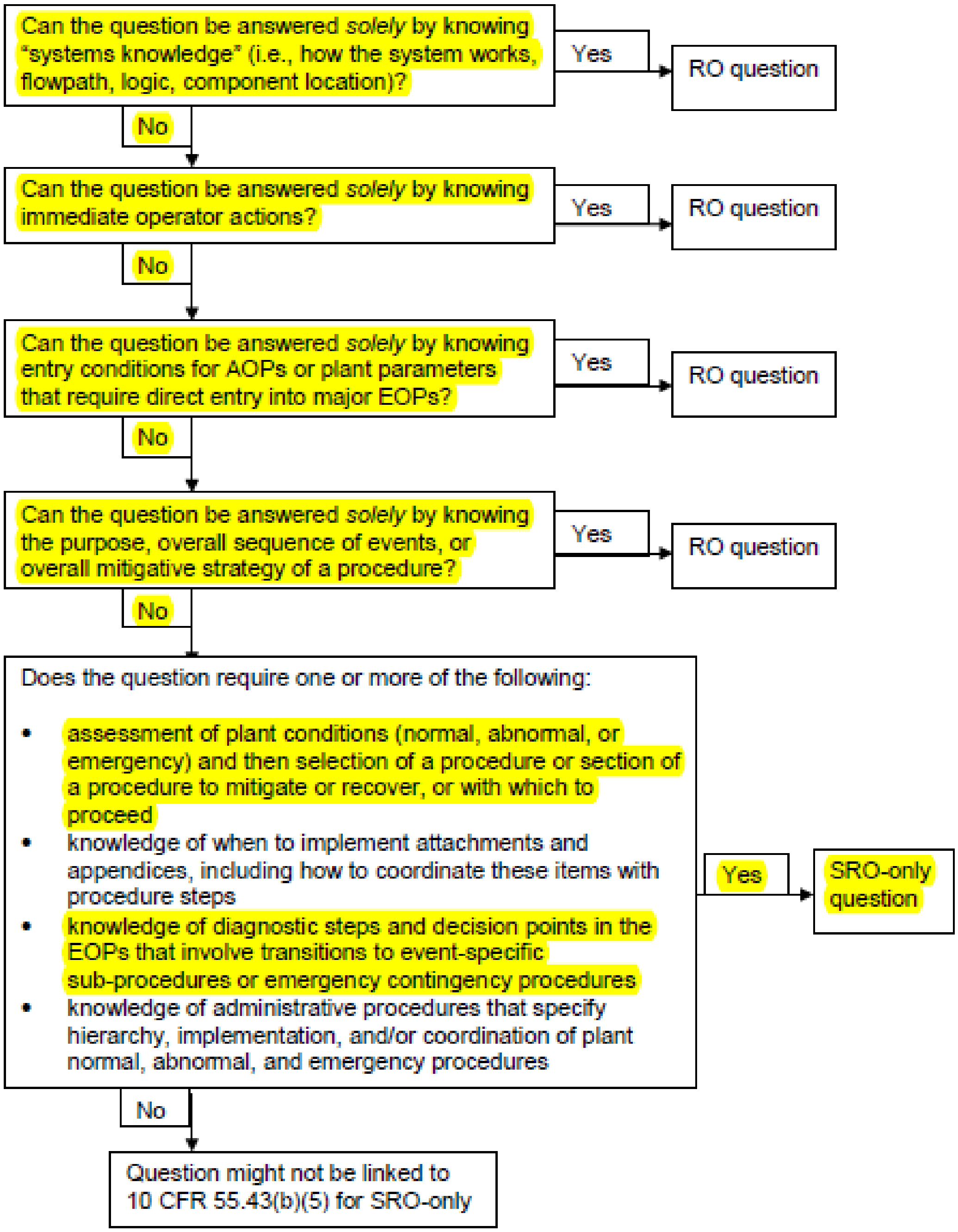
Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		**Memory or Fundamental Knowledge**
	X	**Comprehension or Analysis**

Level of Difficultly:	**3**	
10CFR55.43:	**5**	
Reference Provided:	**N**	
Learning Objective:	**Given conditions of LOCA, analyze RCS Inventory Control to determine if the SFSC acceptance criteria is satisfied per 40EP-9EO03.**	

Technical Reference:	NUREG 1021 SRO-Only Guidance

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Technical Reference:	Technical Specifications (LCO 3.5.3.B bases)

B.1

If one or more ECCS trains are inoperable, except for reasons other than Condition A (one LPSI subsystem inoperable), and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available, the inoperable components must be returned to OPERABLE status within 72 hours. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. The 72 hour Completion Time is based on an NRC study (Ref. 4) using a reliability evaluation and is a reasonable amount of time to effect many repairs.

An ECCS train is inoperable if it is not capable of delivering the design flow to the RCS. The individual components are inoperable if they are not capable of performing their design function, or if supporting systems are not available.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of OPERABLE equipment such that 100% of the ECCS flow equivalent to 100% of a single OPERABLE train remains available. This allows increased flexibility in plant operations when components in opposite trains are inoperable.

An event accompanied by a loss of offsite power and the failure of an emergency DG can disable one ECCS train until power is restored. A reliability analysis (Ref. 4) has shown that the impact with one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

With one or more components inoperable, such that 100% of the equivalent flow to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

Technical Reference:	UFSAR

EMERGENCY CORE COOLING SYSTEM

6.3.2.5.4 Capacity to Maintain Cooling Following a Single Failure.

Minimum operability requirements for components of the ECCS are as delineated in PVNGS Technical Specifications, Section 3.5. Consistent with these operability requirements and system failure modes, the minimum ECCS equipment that will operate during postulated accidents is as discussed in Section 6.3.3. This complement of equipment is required to mitigate the consequences of a LOCA initiated when the reactor is anywhere from hot shutdown to full power operation, and this complement will result in conservative results for other incidents where ECCS is required.

Technical Reference:	40EP-9EO03 LOCA

One HPSI pump and one LPSI pump will provide adequate SI flow during an accident. LOCA EOP SFSCs will be met and entry into the FRP will not be required.

1. Monitor the SFSCs by performing the following: a. Check that the Safety Function Status Check acceptance criteria are satisfied. b. Ensure that the Steam Generator Sample Valves are open. c. Direct Chemistry to PERFORM 74DP-9ZZ05, Abnormal Occurrence Checklist	1.1 Perform the following: a. Rediagnose the event. b. GO TO **ONE** of the following: • Appropriate Optimal Recovery Procedure • 40EP-9EO09, Functional Recovery
* 5. **IF** SIAS has actuated, **THEN** perform the following:	
a. Check that the HPSI and LPSI Pumps have started.	a.1 Start idle HPSI and LPSI Pumps as necessary.
b. Check that safety injection flow is adequate. REFER TO Appendix 2, Figures.	b.1 Perform the following: 1) Ensure electrical power to valves and pumps. 2) Ensure correct control board valve lineup. 3) Ensure operation of ESF auxiliary systems. 4) Start idle charging pumps as needed.

RCS Inventory Control

---------- **NOTE** ----------

Meeting the provisions of Condition 1 or Condition 2 wil Control Safety Function.

ACCEPTANCE CRITERIA:

Condition 1

a. Pressurizer level greater than 10% [15%].

b. RCS is 24°F [44°F] or more subcooled.

c. RVLMS indicates that RVUH level is 16% or more.

Condition 2

a. Safety Injection flow is adequate. REFER TO Appendix 2, Figures.

b. CET Subcooling indicates less than 44°F [60°F] superheat and **NOT** rising.

c. RCS Subcooling indicates less than 44°F [60°F] superheat and **NOT** rising.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Large Break LOCA: Ability to determine or interpret the following as they apply to a Large Break LOCA: Difference between overcooling and LOCA indications	**Tier**			**1**
	Group			**1**
	K/A		**011 EA2.13**	
	IR			**3.7**

Question 77

Given the following conditions:

- Unit 2 was tripped due to a high energy release inside Containment
- SPTAs have been completed and the CRS has entered the optimal EOP
- Current plant conditions are as follows:
  - SIAS/CIAS/MSIS/CSAS have all actuated
  - All RCPs are secured
  - Thot is 545°F and slowly lowering
  - Tcold is 540°F and slowly lowering
  - REPCET is 550°F and slowly lowering
  - RCS pressure is 1050 psia and slowly lowering
  - Containment Radiation Monitor readings are slowly rising
  - Steam Plant Radiation Monitor readings are stable
  - Containment pressure is 12 psig and slowly lowering

Per the appropriate EOP for this condition, the Core Heat Removal Safety Function is currently ___(1)___ and the Containment Isolation Safety Function is currently ___(2)___ .

A. (1) MET
   (2) MET

B. (1) MET
   (2) NOT met

C. (1) NOT met
   (2) MET

D. (1) NOT met
   (2) NOT met

Proposed Answer:	A

Explanations:	
A.	Correct. The correct EOP for this condition is LOCA (based on the loss of subcooling and the containment RM trends) and both safety functions are currently met.
B.	First part is correct. Second part is plausible since Condition 1 is not met due to containment pressure and containment RM trends, however condition 2 is met for the LOCA EOP.
C.	First part is plausible since subcooling is lost (RCS is ~ 0°F subcooled), however for the LOCA EOP, the safety function is met if CET subcooling indicates less than 44°F superheated. Second part is correct.
D.	First part is plausible since subcooling is lost (RCS is ~ 0°F subcooled), however for the LOCA EOP, the safety function is met if CET subcooling indicates less than 44°F superheated. Second part is plausible since Condition 1 is not met due to containment pressure and containment RM trends, however condition 2 is met for the LOCA EOP.

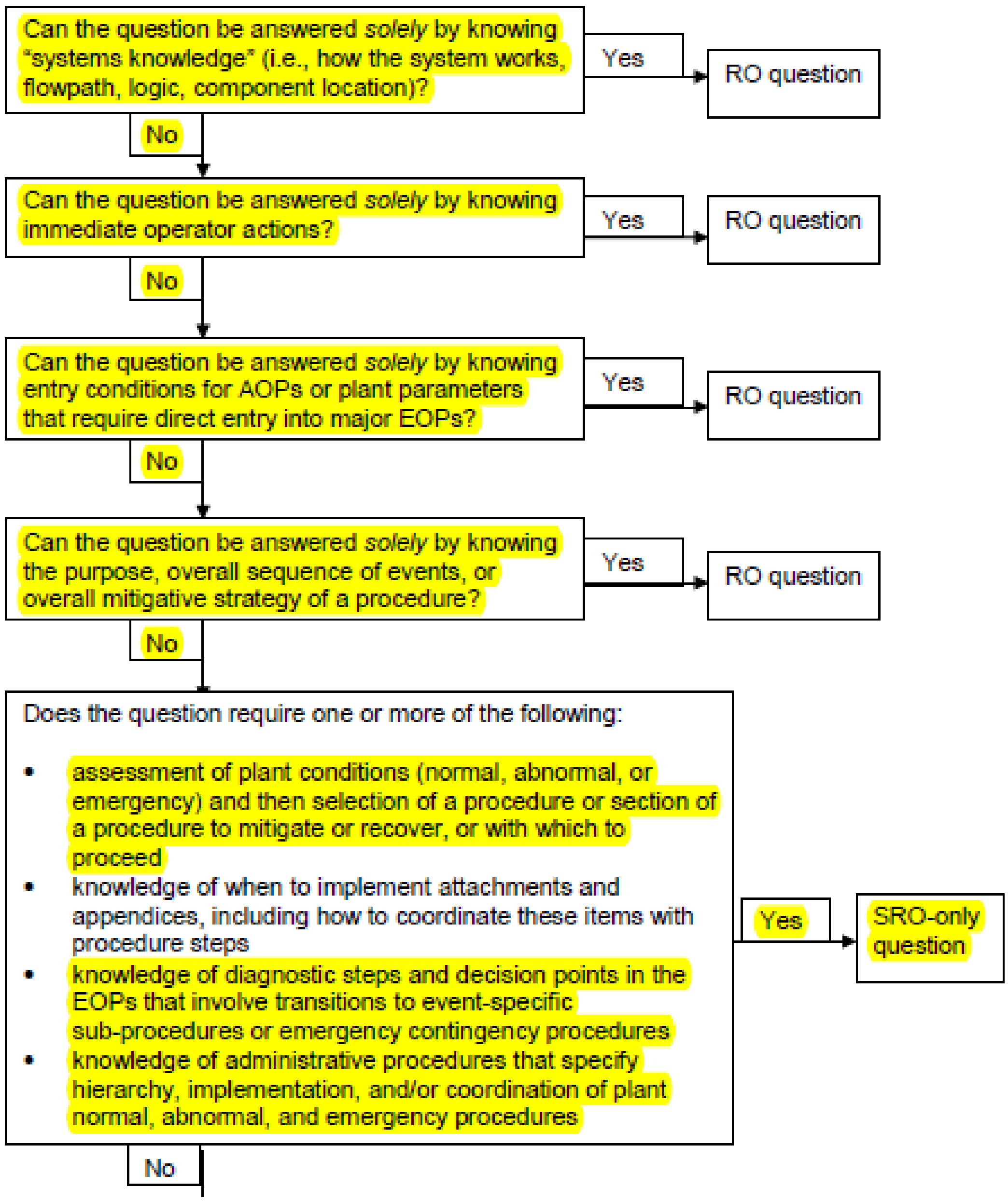
Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		**Memory or Fundamental Knowledge**
	X	**Comprehension or Analysis**

Level of Difficultly:	**3**	
10CFR55.43:	**5**	
Reference Provided:	**Y**	**Steam Tables (to determine subcooling)**
Learning Objective:	**Given conditions of LOCA, analyze Core Heat Removal to determine if the SFSC acceptance criteria is satisfied per 40EP-9EO03.**	

Technical Reference:	NUREG 1021 SRO-Only Guidance

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



Technical Reference:	40EP-9EO03, LOCA

SAFETY FUNCTION:

5. Core Heat Removal

ACCEPTANCE CRITERIA:	CRITERIA SATISFIED
a. CET Subcooling indicates less than 44°F [60°F] superheat and **NOT** rising.	☐ ☐ ☐ ☐
b. RCS Subcooling indicates less than 44°F [60°F] superheat and **NOT** rising.	☐ ☐ ☐ ☐

Technical Reference:	40EP-9EO03, LOCA

SAFETY FUNCTION:

7. Containment Isolation

---------- **NOTE** ----------

Meeting the provisions of Condition 1 or Condition 2 will satisfy the Containment Isolation Safety Function.

ACCEPTANCE CRITERIA:	CRITERIA SATISFIED
Condition 1	
a. No valid steam plant activity radiation monitor alarms or unexplained rise in activity.	☐ ☐ ☐ ☐
b. Containment pressure is less than 3 psig.	☐ ☐ ☐ ☐
c. No valid containment area radiation monitor alarms or unexplained rise in activity.	☐ ☐ ☐ ☐
Condition 2	
a. No valid steam plant activity radiation monitor alarms or unexplained rise in activity.	☐ ☐ ☐ ☐
b. CIAS actuated.	☐ ☐ ☐ ☐

Technical Reference:	40EP-9EO05, ESD

SAFETY FUNCTION:

5. Core Heat Removal

ACCEPTANCE CRITERIA:	CRITERIA SATISFIED
a. T_h is less than 650°F [650°F].	☐ ☐ ☐ ☐
b. Maximum quadrant CET temperature is less than 650°F [650°F].	☐ ☐ ☐ ☐
c. The RCS is 24°F [44°F] or more subcooled.	☐ ☐ ☐ ☐

Technical Reference:	40EP-9EO05, ESD

SAFETY FUNCTION:

7. Containment Isolation

ACCEPTANCE CRITERIA:	CRITERIA SATISFIED
a. Containment pressure is less than 3 psig. **OR** CIAS actuated.	☐ ☐ ☐ ☐
b. No valid containment area radiation monitor alarms or unexplained rise in activity.	☐ ☐ ☐ ☐
c. No valid steam plant activity radiation monitor alarms or unexplained rise in activity. **OR** **IF** radiation monitors and steam generator samples are **NOT** available, **THEN** radiation and contamination surveys of Steam Generator release points show no unexplained rise in steam plant activity.	☐ ☐ ☐ ☐

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Reactor Coolant Pump Malfunctions: Knowledge of limiting conditions for operations and safety limits	**Tier**		[redacted]	**1**
	Group		[redacted]	**1**
	K/A	**015 G 2.2.22**		
	IR		[redacted]	**4.7**

Question 78

(1) Per the PVNGS Technical Specification Bases, the RPS function which is designed to protect the core in the event of an RCP SHEARED shaft is…

(2) Per Technical Specifications, during a Reactor startup, the EARLIEST time that the function described in Part 1 is required to be OPERABLE is when…

A. (1) Reactor Coolant Flow (SG D/P) – Low
(2) MODE 2 is entered

B. (1) Reactor Coolant Flow (SG D/P) – Low
(2) any RTCBs are closed and any CEA is capable of being withdrawn

C. (1) Departure From Nucleate Boiling Ratio – Low
(2) MODE 2 is entered

D. (1) Departure From Nucleate Boiling Ratio – Low
(2) any RTCBs are closed and any CEA is capable of being withdrawn

Proposed Answer:	A

Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible since this is earliest time in a startup that any RPS trip is required to be operable, however RC Flow Low is not required to be operable until MODE 2 is entered.
C.	First part is plausible since DNBR Low is the RPS trip which will trip the Reactor during most lost of RC Flow events, however since the DNBR trip is generated from RCP motor speed, it will not detect a sheared shaft. Second part is correct.
D.	First part is plausible since DNBR Low is the RPS trip which will trip the Reactor during most lost of RC Flow events, however since the DNBR trip is generated from RCP motor speed, it will not detect a sheared shaft. Second part is plausible since this is earliest time in a startup that any RPS trip is required to be operable, however RC Flow Low is not required to be operable until MODE 2 is entered.

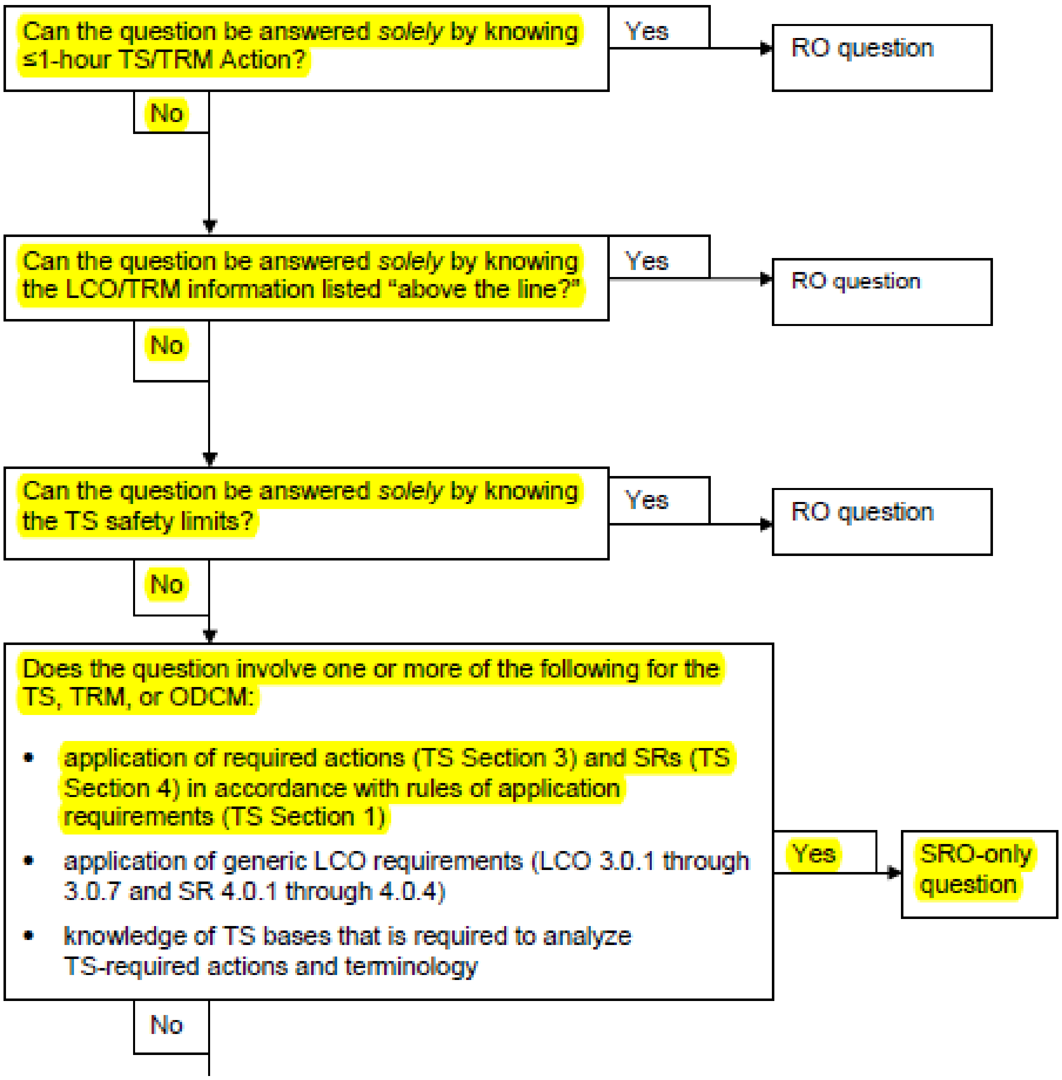
Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	2	
Reference Provided:	N	
Learning Objective:	Given a set of plant conditions, determine whether or not the LCOs and TLCOs of 3.3 are satisfied in accordance with Tech Spec 3.3.	

Technical Reference:	NUREG 1021 SRO-Only Guidance

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Technical Reference:	Tech Spec Bases

BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

8, 9. Steam Generator Level - Low

The Steam Generator #1 Level - Low and Steam Generator #2 Level - Low trips ensure that a reactor trip signal is generated for the following events to help prevent exceeding the design pressure of the RCS due to the loss of the heat sink:

- Inadvertent Opening of a Steam Generator Atmospheric Dump Valve (AOO);
- Loss of Condenser Vacuum (AOO);
- Loss of Normal Feedwater Event (AOO);
- Feedwater System Pipe Break (Accident); and
- Single RCP Rotor Seizure (AOO)

10, 11. Steam Generator Level - High

The Steam Generator #1 Level - High and Steam Generator #2 Level - High trips are provided to protect the turbine from excessive moisture carryover in case of a steam generator overfill event. A Main Steam Isolation Signal (MSIS) is initiated simultaneously.

12, 13. Reactor Coolant Flow - Low

The Reactor Coolant Flow Steam Generator #1-Low and Reactor Coolant Flow Steam Generator #2-Low trips provide protection against an RCP Sheared Shaft Event. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a variable setpoint. This variable setpoint stays below the pressure differential by a preset value called the step function, unless limited by a preset maximum decreasing rate determined by the Ramp Function, or a set minimum value determined by the Floor Function. The setpoints ensure that a reactor trip occurs to limit fuel failure and ensure offsite doses are within 10 CFR 100 guidelines.

Technical Reference:	Tech Spec Bases

BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

15. Departure from Nucleate Boiling Ratio (DNBR) - Low

The CPCs perform the calculations required to derive the DNBR and LPD parameters and their associated RPS trips. The DNBR - Low and LPD - High trips provide plant protection during the following AOOs and assist the ESF systems in the mitigation of the following accidents.

The DNBR - Low trip provides protection against core damage due to the occurrence of locally saturated conditions in the limiting (hot) channel during the following events and is the primary reactor trip (trips the reactor first) for these events:

- Decrease in Feedwater Temperature;
- Increase in Feedwater Flow;
- Increased Main Steam Flow (not due to steam line rupture) Without Turbine Trip;
- Increased Main Steam Flow (not due to steam line rupture) With a Concurrent Single Failure of an Active Component;
- Steam Line Break With Concurrent Loss of Offsite AC Power;
- Loss of Normal AC Power;
- Partial Loss of Forced Reactor Coolant Flow;
- Total Loss of Forced Reactor Coolant Flow;
- Single Reactor Coolant Pump (RCP) Shaft Seizure;
- Uncontrolled CEA Withdrawal From Low Power;
- Uncontrolled CEA Withdrawal at Power;
- CEA Misoperation, except for dropped 4-finger CEA event;
- Primary Sample or Instrument Line Break; and
- Steam Generator Tube Rupture.

Technical Reference:	Tech Specs

Table 3.3.1-1 (page 2 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
8. Steam Generator #1 Level - Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≥ 43.7%
9. Steam Generator #2 Level -Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≥ 43.7%
10. Steam Generator #1 Level - High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 91.5%
11. Steam Generator #2 Level - High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 91.5%
12. Reactor Coolant Flow, Steam Generator #1-Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Ramp: ≤ 0.115 psid/sec. Floor: ≥ 12.49 psid Step: ≤ 17.2 psid
13. Reactor Coolant Flow, Steam Generator #2-Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Ramp: ≤ 0.115 psid/sec. Floor: ≥ 12.49 psid Step: ≤ 17.2 psid

Technical Reference:	Tech Specs

Table 3.3.1-1 (page 3 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
14. Local Power Density - High[b]	1,2	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.11 SR 3.3.1.12 SR 3.3.1.13	≤ 21.0 kW/ft
15. Departure From Nucleate Boiling Ratio (DNBR) - Low[b]	1,2	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.11 SR 3.3.1.12 SR 3.3.1.13	≥ 1.34

Technical Reference:	Tech Specs

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALVE
1. Logarithmic Power Level-High[d]	3[a], 4[a], 5[a]	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.4 SR 3.3.2.5	≤ 0.011% NRTP[c]
2. Steam Generator #1 Pressure-Low[b]	3[a]	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	≥ 955 psia[e]
3. Steam Generator #2 Pressure-Low[b]	3[a]	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	≥ 955 psia[e]

(a) With any Reactor Trip Circuit Breakers (RTCBs) closed and any control element assembly capable of being withdrawn.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Reactor Coolant Makeup: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions	**Tier**			**1**
	Group			**1**
	K/A	**022 G 2.2.44**		
	IR			**4.4**

Question 79

Given the following conditions:

- Unit 1 is operating at 100% power
- The ‘A’ and ‘B’ Charging Pumps are running
- VCT level is 40%
- The following alarms have just annunciated on B03:
  - 3A08A CHG HDR SYS TRBL
  - 3A11B RCP SEAL INJ FLOW HI-HI OR LO
- CHB-FI-212, Charging Pumps Discharge Header Flow, is indicating 25 gpm
- The CRS has entered 40AO-9ZZ05, Loss of Charging or Letdown, Appendix G, Responding to Gas Binding of Charging Pumps

Per 40AO-9ZZ05, Loss of Charging or Letdown, the CRS should direct…

(1) the OATC to place _____ in Pull to Lock

(2) an AO to perform _____ to vent the affected Charging Pumps

A. (1) ONLY the ‘A’ and ‘B’ Charging Pumps
(2) Appendix H, Venting Charging Pumps and Header to the Vent Receiver

B. (1) ONLY the ‘A’ and ‘B’ Charging Pumps
(2) Appendix I, Venting Charging Pumps and Header to the Recycle Drain Header

C. (1) ALL three Charging Pumps
(2) Appendix H, Venting Charging Pumps and Header to the Vent Receiver

D. (1) ALL three Charging Pumps
(2) Appendix I, Venting Charging Pumps and Header to the Recycle Drain Header

Proposed Answer:	D

Explanations:	
A.	First part is plausible since only the ‘A’ and ‘B’ Charging Pumps were running, and maintaining the ‘E’ Charging Pump available leaves the option to restore letdown and seal injection, however if Charging discharge flow is < 40 gpm, placing all three pumps in pull to lock is required. Second part is plausible since Appendix H is used to vent the charging pumps and header following a gas binding event, however only when VCT level is 0%.
B.	First part is plausible since only the ‘A’ and ‘B’ Charging Pumps were running, and maintaining the ‘E’ Charging Pump available leaves the option to restore letdown and seal injection, however if Charging discharge flow is < 40 gpm, placing all three pumps in pull to lock is required. Second part is correct.
C.	First part is correct. Second part is plausible since Appendix H is used to vent the charging pumps and header following a gas binding event, however only when VCT level is 0%.
D.	Correct.

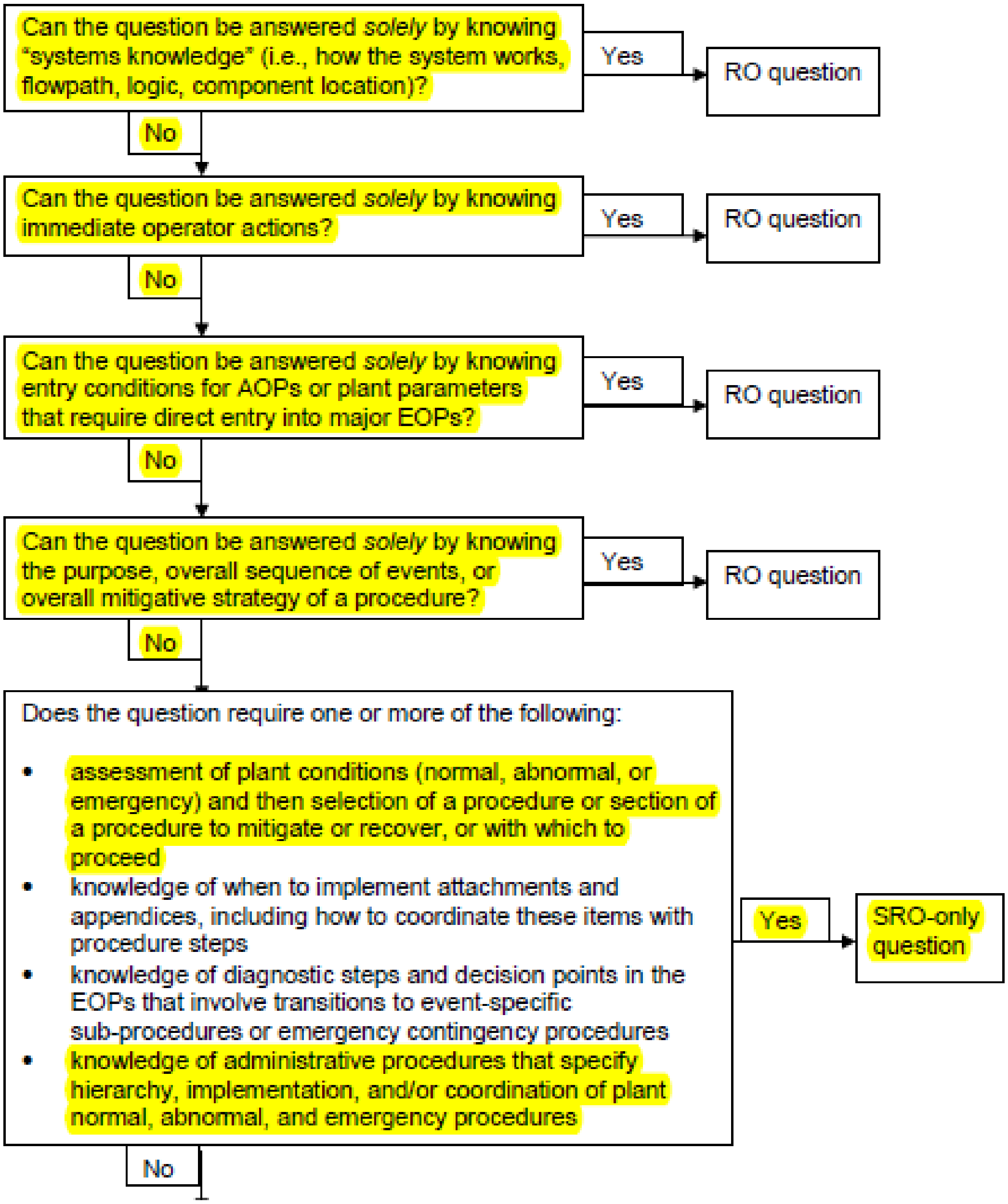
Question Source:	x	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	x	Comprehension or Analysis

Level of Difficultly:	4	
10CFR55.43:	5	
Reference Provided:	N	
Learning Objective:	Explain how gas binding of the charging pumps is mitigated in 40AO-9ZZ05, Loss of Charging or Letdown	

Technical Reference:	NUREG 1021 SRO-Only Guidance

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Technical Reference:	40AO-9ZZ05 Loss of Charging or Letdown

INSTRUCTIONS CONTINGENCY ACTIONS

NOTE

Multiple indications and SM/CRS discretion should be applied to diagnosing Charging Pump gas binding.

4. **IF** Charging Pump gas binding is indicated by **ANY** of the following:
   - Charging header flow fluctuations
   - Charging header pressure fluctuations
   - Charging header flow less than expected for running charging pumps
   - Charging suction source (VCT, RWT) level lost

   THEN PERFORM Appendix G, Responding to Gas Binding of Charging Pumps.

Step 4 will be performed because charging flow is < 40 gpm. (PTL all charging pumps)

Appendix G, Responding to Gas Binding of Charging Pumps

INSTRUCTIONS CONTINGENCY ACTIONS

1. Enter Appendix Entry Time and Date:

   ____

2. **IF** CHB-FI-212, Charging Pump to Regen HX, indicates greater than 40 gpm,
   THEN GO TO Step 8.

3. Close CHB-UV-515, Letdown To Regen HX Isolation Valve, to isolate letdown flow.

4. Place **ALL** of the following handswitches in "PULL TO LOCK":
   - CHA-HS-216, Charging Pump 1 P01
   - CHB-HS-217, Charging Pump 2 P01
   - CHA-HS-218A, Charging Pump 3 P01
   - CHB-HS-218, Charging Pump 3 P01

5. **IF** gas intrusion was due to VCT level lowering below 0%,
   THEN PERFORM Appendix H, Venting Charging Pumps and Header to the Vent Receiver.

VCT level is 40%, which makes step 5 not applicable. Venting to the vent receiver will NOT be performed.

Technical Reference:	40AO-9ZZ05 Loss of Charging or Letdown

Step 6 is applicable because VCT did NOT go below 0%. Venting to the Recycle Drain Header will be performed.

Appendix G, Responding to Gas Binding of Charging Pumps

INSTRUCTIONS | CONTINGENCY ACTIONS

_ 6. **IF** gas intrusion was **NOT** due to VCT level lowering below 0%, **THEN** PERFORM Appendix I, Venting Charging Pumps and Header to the Recycle Drain Header.

_ 7. GO TO Section 3.0, Step 5 **OR** Section 4.0, Step 4.

-- **NOTE** --

With two charging pumps operating while one of the pumps is gas bound, the primary indication of the gas bound pump will be the sound. A charging pump that is partly gas bound will initially have much louder cavitation noises than a filled pump. As the pump becomes fully gas bound, the plate valves will make much less noise than those in a pump that is filled with fluid.

_ 8. Determine which Charging Pump has been gas bound using local observation.

_ 9. Place the handswitch for the gas bound Charging Pump in "PULL TO LOCK":

- CHA-HS-216, Charging Pump 1 P01
- CHB-HS-217, Charging Pump 2 P01
- CHA-HS-218A, Charging Pump 3 P01
- CHB-HS-218, Charging Pump 3 P01

Step 8 and 9 will only be performed is charging flow was > 40 gpm (refer to previous step 2). These steps would not be performed in this case because we kick out at step 7.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Steam Generator Tube Rupture: Ability to determine or interpret the following as they apply to a SGTR: Magnitude of atmospheric radioactive release if cooldown must be completed using steam dumps or if atmospheric reliefs lift	**Tier**			**1**
	Group			**1**
	K/A	**038 EA2.14**		
	IR			**4.6**

Question 80

Given the following conditions:

- Unit 1 was tripped due to a Design Basis Steam Generator Tube Rupture event on SG #1
- On the trip, offsite power was lost
- The crew is commencing a cooldown using ADVs to meet conditions required to isolate SG #1

(1) Per the PVNGS EAL Hot Chart, the use of ADVs for the INITIAL cooldown _____ considered a loss of the Containment Barrier.

(2) Per the PVNGS Release Evaluation Flowchart, the release in progress _____ exceeding federally approved limits.

A. (1) IS
(2) IS

B. (1) IS
(2) is NOT

C. (1) is NOT
(2) IS

D. (1) is NOT
(2) is NOT

Proposed Answer:	D

Explanations:	
A.	First part is plausible because during the initial RCS cooldown, there will be a release to the environment, however since it is not an unisolable fault (e.g. stuck open MSSV), this is NOT considered a loss of the containment barrier. Second part is plausible because for the initial cooldown, there will be a release to the environment, however per EP-0903, Accident Assessment and the Release Evaluation Flowchart, this is not a release that exceeds federal limits.
B.	First part is plausible because during the initial RCS cooldown, there will be a release to the environment, however since it is not an unisolable fault (e.g. stuck open MSSV), this is NOT considered a loss of the containment barrier. Second part is correct.
C.	First part is correct. Second part is plausible because for the initial cooldown, there will be a release to the environment, however per EP-0903, Accident Assessment and the Release Evaluation Flowchart, this is not a release that exceeds federal limits.
D.	Correct.

Question Source:		New	
	X	Bank	
		Modified	
	X	Previous NRC Exam	2020 NRC Q85

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	4	
Reference Provided:	N	
Learning Objective:	Determine whether a radioactive release is in progress	

Technical Reference:	NUREG 1021 SRO-Only Guidance

D. Radiation Hazards That May Arise during Normal and Abnormal Situations, including Maintenance Activities and Various Contamination Conditions [10 CFR 55.43(b)(4)]

Some examples of SRO exam items for this topic include the following:

- process for gaseous/liquid release approvals (i.e., release permits)
- analysis and interpretation of radiation and activity readings as they pertain to the selection of administrative, normal, abnormal, and emergency procedures
- analysis and interpretation of coolant activity, including comparison to emergency plan criteria and/or regulatory limits

Technical Reference:	EAL Hot Chart (Fission Product Barriers)

A ruptured SG that is being steamed to atmosphere is evaluated for a Loss of CTMT barrier.

Containment (CTMT) Barrier	
Loss	**Potential Loss**
1. A leaking or RUPTURED SG is FAULTED outside of containment	

Technical Reference:	PVNGS Emergency Plan

If a ruptured SG is steamed to atmosphere (i.e. manual operation of ADVs), it could potentially meet the loss of CTMT barrier threshold. For the initial cooldown to 540F, use of the ADVs is assumed in the accident analysis and would not meet the threshold. However, if the ruptured SG has to be steamed below 540F it would be a loss of the CTMT barrier.

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

1. A leaking or RUPTURED SG is FAULTED outside of containment

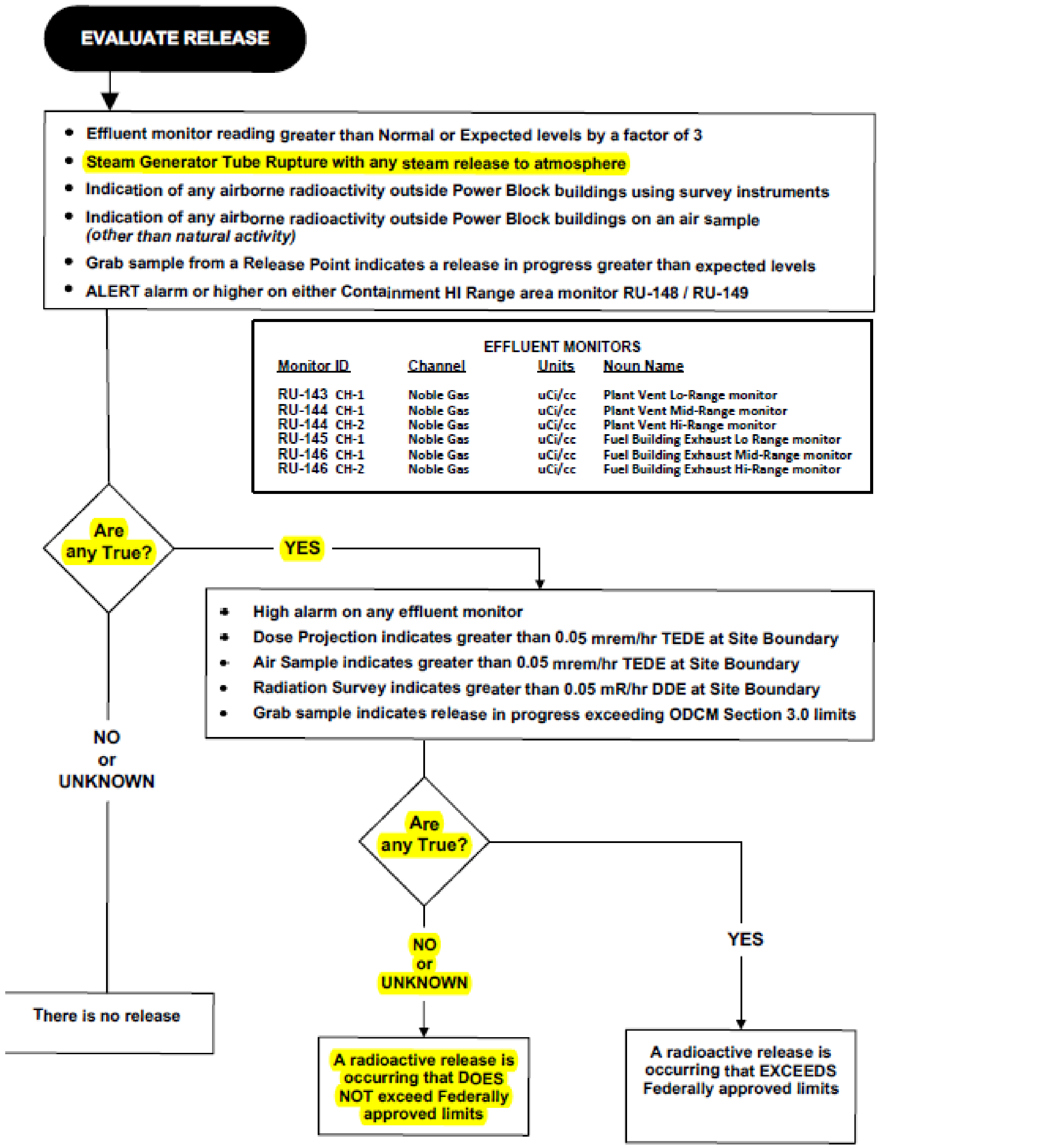
This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking or RUPTURED steam generator directly to atmosphere to cooldown the plant. These type of condition will result in a significant and sustained release of radioactive steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow without an adverse effect on plant cooldown meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG Atmospheric Dump Valve(s) do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. This includes the initial cooldown to 540°F to isolate the ruptured SG using Atmospheric Dump Valves directed in the SGTR EOP. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.

Technical Reference:	EP-0903 Release Evaluation Flowchart

Design bases SGTR will not result in 10CFR100 limits being exceeded. Effluent rad monitor alarms would not be expected. Dose assessments would not yet be available during the initial SGTR cooldown and they would not be expected to be exceeding the limits. A release is in progress that is within federally approved limits.

RELEASE EVALUATION FLOWCHART



EFFLUENT MONITORS

Monitor ID	Channel		Units	Noun Name
RU-143	CH-1	Noble Gas	uCi/cc	Plant Vent Lo-Range monitor
RU-144	CH-1	Noble Gas	uCi/cc	Plant Vent Mid-Range monitor
RU-144	CH-2	Noble Gas	uCi/cc	Plant Vent Hi-Range monitor
RU-145	CH-1	Noble Gas	uCi/cc	Fuel Building Exhaust Lo Range monitor
RU-146	CH-1	Noble Gas	uCi/cc	Fuel Building Exhaust Mid-Range monitor
RU-146	CH-2	Noble Gas	uCi/cc	Fuel Building Exhaust Hi-Range monitor

Examination Outline Cross-Reference:	**Level**	**RO**	[redacted]	**SRO**
K/A: Loss of DC Power: Ability to determine and interpret the following as they apply to the Loss of DC Power: 125V dc bus voltage, low/critical low, alarm	**Tier**		[redacted]	**1**
	Group			**1**
	K/A	**058 AA2.02**		
	IR		[redacted]	**3.6**

Question 81

Given the following conditions:

- Unit 3 is operating at 100% power
- All Class 125 VDC components are operable
- Both Swing Chargers are in standby

Subsequently:

- **At time = 0100:** ‘A’ Battery Charger, PKA-H11, failed and has no output voltage
- **At time = 0115:** ‘A’ Battery, PKA-F11, output voltage dropped below the minimum required voltage for operability
- **At time = 0130:** ‘AC’ Swing Charger, PKA-H15, was aligned to PKA-M41
- **At time = 0200:** ‘A’ Battery, PKA-F11, output voltage was restored to minimum required voltage for operability

Based on the listed timeline of events, LCO 3.8.4, DC Sources – Operating, was INITIALLY NOT MET at ___(1)___ , and was subsequently MET AS SOON AS ___(2)___ .

A. (1) 0100
(2) the ‘AC’ Swing Charger was aligned to PKA-M41

B. (1) 0100
(2) ‘A’ Battery voltage was restored to minimum required voltage

C. (1) 0115
(2) the ‘AC’ Swing Charger was aligned to PKA-M41

D. (1) 0115
(2) ‘A’ Battery voltage was restored to minimum required voltage

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible since power is restored to the bus when the swing charger is aligned, however operability restoration requires battery voltage to be in spec as well.
B.	Correct.
C.	First part is plausible since voltage on PKA-M41 is still fine based on being powered by the battery until voltage is too low at 0115, however per LCO 3.8.4, a charger must be aligned to each PK bus to be considered operable. Second part is plausible since power is restored to the bus when the swing charger is aligned, however operability restoration requires battery voltage to be in spec as well.
D.	First part is plausible since voltage on PKA-M41 is still fine based on being powered by the battery until voltage is too low at 0115, however per LCO 3.8.4, a charger must be aligned to each PK bus to be considered operable. Second part is correct.

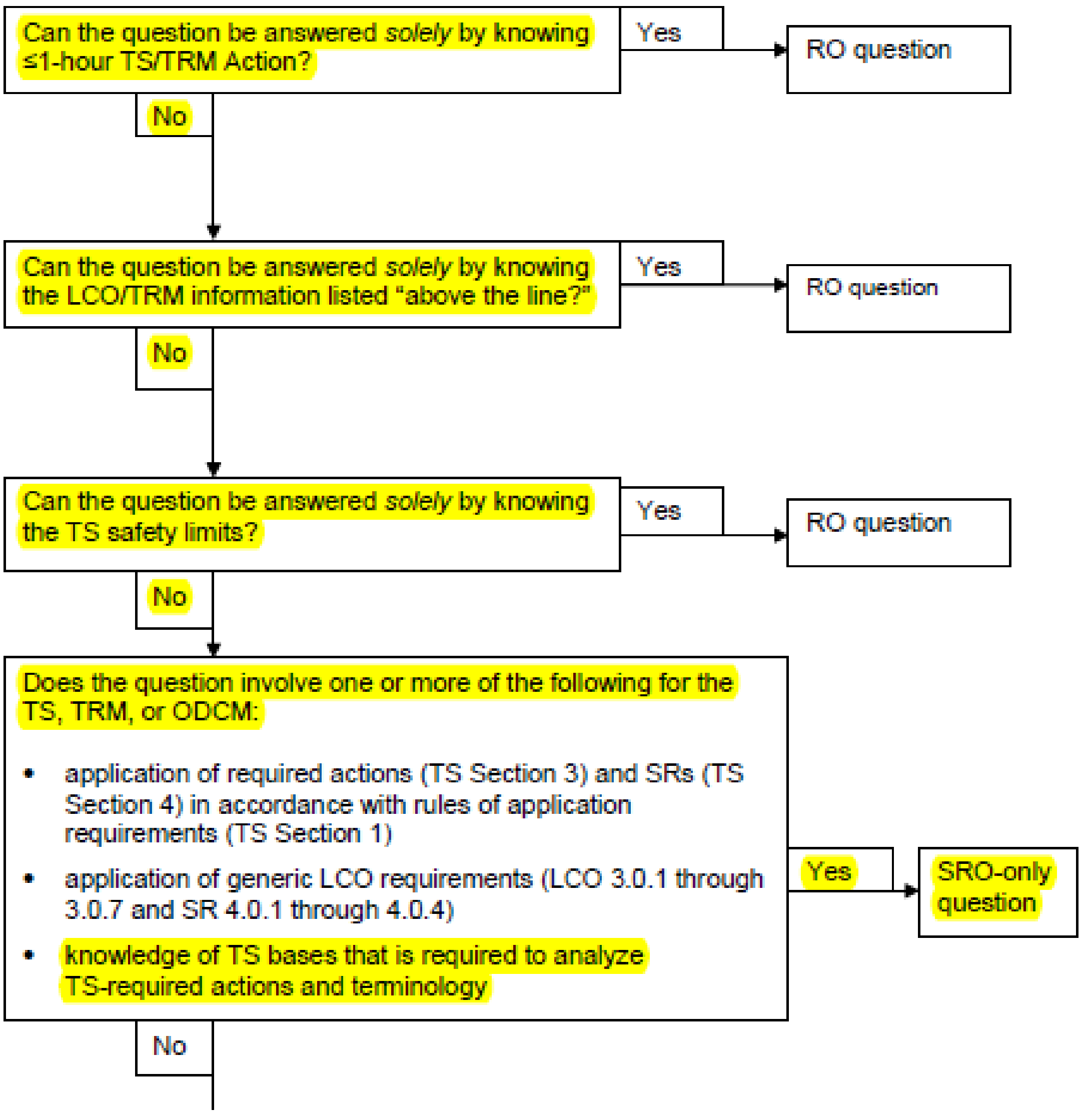
Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	4	
10CFR55.43:	2	
Reference Provided:	N	
Learning Objective:	Given a set of plant conditions, determine whether or not the LCOs and TLCOs of 3.8 are satisfied in accordance with Tech Spec 3.8.	

Technical Reference:	NUREG 1021 SRO-Only Guidance

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Technical Reference:	**Technical Specifications**

When the battery charger fails, LCO 3.8.4.A is entered (0100). Additionally, not meeting battery terminal voltage results in the LCO not being met. This is due to not meeting SR 3.8.4.1.

To exit the Condition and to meet LCO 3.8.4, a charger must be restored AND battery terminal voltage must be met.

LCO 3.8.4 The Train A and Train B DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One battery charger on one subsystem inoperable.	A.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage. <u>AND</u> A.2 Verify battery float current ≤ 2 amps. <u>AND</u> A.3 Restore battery charger to OPERABLE status.	2 hours Once per 12 hours 72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. One DC electrical power subsystem inoperable for reasons other than Condition A.	B.1 Restore DC electrical power subsystem to OPERABLE status.	2 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is greater than or equal to the minimum established float voltage.	In accordance with the Surveillance Frequency Control Program

Technical Reference:	Technical Specifications Bases

LCO

The DC electrical power subsystems, each subsystem consisting of two batteries, battery charger for each battery (the backup battery charger, one per train, may be used to satisfy this requirement), and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the subsystem are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. Loss of any DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 4).

Each DC electrical power subsystem (Train A or Train B) is subdivided into channels. Train A consists of Channel A and Channel C. Train B consists of Channel B and Channel D. Channel A includes 125 VDC bus PKA-M41, 125 VDC battery bank PKA-F11, and normal battery charger PKA-H11 or backup battery charger PKA-H15. Channel C includes 125 VDC bus PKC-M43, 125 VDC battery bank PKC-F13, and normal battery charger PKC-H13 or backup battery charger PKA-H15.

Channel B includes 125 VDC bus PKB-M42, 125 VDC battery bank PKB-F12, and normal battery charger PKB-H12 or backup battery charger PKB-H16. Channel D includes 125 VDC bus PKD-M44, 125 VDC battery bank PKD-F14, and normal battery charger PKD-H14 or backup battery charger PKB-H16.

An OPERABLE DC electrical power subsystem requires all required batteries and respective chargers to be operating and connected to the associated DC bus(es).

ACTIONS

A.1, A.2, and A.3

Condition A represents one subsystem with one battery charger inoperable (e.g., the voltage limit of SR 3.8.4.1 is not maintained). The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action A.1 requires that the battery terminal voltage be restored to greater than or equal to the minimum established float voltage (2.17 volts per cell (Vpc) times the number of connected cells or 130.2 V for a 60 cell battery at the battery terminals) within 2 hours. This time provides for returning the inoperable charger to OPERABLE status or providing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within 12 hours, the battery will be restored to its fully charged condition (Required Action A.2) from fully charged condition any discharge that might have occurred due to the charger inoperability.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Continuous Rod Withdrawal: Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal : Proper actions to be taken if automatic safety functions have not taken place	Tier			1
	Group			2
	K/A	001 AA2.03		
	IR			4.8

Question 82

Given the following conditions:

- Unit 2 is operating at 14% power during a power ascension
- Group 5 CEAs are 125” withdrawn
- The OATC is withdrawing Group 5 CEAs to achieve swapover
- The CEA Withdrawal Switch was released when Group 5 CEAs were at 128”
- After releasing the switch, Group 5 CEAs continued to withdraw
- All actions to stop the CEA withdrawal failed
- Prior to attempting to manually trip the Reactor, the OATC observed valid Reactor trip signals on all 4 CPCs
- The OATC depressed all 4 RTCB pushbuttons on B05 and the Reactor did NOT trip

(1) The NEXT action the crew should take is to…

(2) Assuming the action taken in part 1 was successful in shutting down the Reactor, the SM should classify the event as an…

A. (1) dispatch an AO to locally open RTCBs
   (2) Unusual Event

B. (1) dispatch an AO to locally open RTCBs
   (2) Alert

C. (1) open NGN-L03 and NGN-L10 feeder breakers
   (2) Unusual Event

D. (1) open NGN-L03 and NGN-L10 feeder breakers
   (2) Alert

Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor	Automatic or manual trip fails to shut down the reactor
SA6.1 1 An automatic or manual trip fails to shut down the reactor as indicated by reactor power > 5% **AND** Manual trip actions taken at the reactor control consoles (B05 or B01) are **not** successful in shutting down the reactor as indicated by reactor power > 5% (Note 8)	**SU6.1** 1 An automatic trip did **not** shut down the reactor as indicated by reactor power > 5% after **any** RPS setpoint is exceeded **AND** A subsequent automatic trip or manual trip action taken at the reactor control consoles (B05 or B01) is successful in shutting down the reactor as indicated by reactor power ≤ 5% (Note 8) **SU6.2** 1 A manual trip did **not** shut down the reactor as indicated by reactor power > 5% after **any** manual trip action was initiated **AND** A subsequent automatic trip or manual trip action taken at the reactor control consoles (B05 or B01) is successful in shutting down the reactor as indicated by reactor power ≤ 5% (Note 8)

Proposed Answer:	C

Explanations:	
A.	First part is plausible since this is a contingency action to trip the Reactor during an ATWS, and it is plausible that it would be preferred to minimize the amount of loads which are de-energized, however local operator of RTCBs is only done if a trip attempt fails at both B05 and B01. Second part is correct.
B.	First part is plausible since this is a contingency action to trip the Reactor during an ATWS, and it is plausible that it would be preferred to minimize the amount of loads which are de-energized, however local operator of RTCBs is only done if a trip attempt fails at both B05 and B01. Second part is plausible since the alert level EAL for an ATWS states, “An automatic or manual trip fails to shut down the reactor as indicated by reactor power > 5% AND manual trip actions taken at the reactor control consoles (B05 or B01) are not successful in shutting down the reactor”, and since an auto trip failed and a manual trip at B05 failed they could think this puts them in an alert, however the alert is only declared if action taken outside the control room is required to trip the reactor.
C.	Correct.
D.	First part is correct. Second part is plausible since the alert level EAL for an ATWS states, “An automatic or manual trip fails to shut down the reactor as indicated by reactor power > 5% AND manual trip actions taken at the reactor control consoles (B05 or B01) are not successful in shutting down the reactor”, and since an auto trip failed and a manual trip at B05 failed they could think this puts them in an alert, however the alert is only declared if action taken outside the control room is required to trip the reactor.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	1	
Reference Provided:	Y	EAL Hot Chart (Only section for ATWS)
Learning Objective:	Determine if an EAL has been met or exceeded	

Technical Reference:	NUREG 1021 SRO-Only Guidance

Having an Emergency Plan is a condition of license, although not specifically called out by the SRO-Only Guidance in NUREG 1021, and implementation of the Emergency Plan is an SRO-Only job function at PVNGS.

A. Conditions and Limitations in the Facility License [10 CFR 55.43(b)(1)]

Examples of SRO exam items for this topic include the following:

- reporting requirements when the maximum licensed thermal power output is exceeded
- administration of fire protection program requirements, such as compensatory actions associated with inoperable sprinkler systems and fire doors
- required actions necessary when a facility does not meet the administrative controls listed in Technical Specifications (TS), Section 5 or 6, depending on the facility (e.g., shift staffing requirements)
- National Pollutant Discharge Elimination System requirements, if applicable
- processes for TS and final safety analysis report changes

:

The SRO-only test item is required to be tied to one of the 10 CFR 55.43(b) items. However, if a licensee desires to evaluate a K/A that is not tied to one of the 10 CFR 55.43(b) items, the licensee can classify the K/A as "unique to the SRO position" provided that there is documented evidence that ties the K/A to the licensee's SRO job position duties in accordance with the systematic approach to training.

Justification. A question that is not tied to one of the 10 CFR 55.43(b) items can still be classified as "SRO-only" provided that the licensee has documented evidence to prove that the K/A is "unique to the SRO position" at the site. An example of documented evidence includes the following:

- The question is linked to a learning objective that is specifically labeled in the lesson plan as being SRO-only (e.g., some facility licensee lesson plans have columns in the margin that differentiate auxiliary operator, RO, and SRO learning objectives). [Section D.2.d of this examination standard]

AND/OR

- A question is linked to a task that is labeled as an SRO-only task, and the task is NOT listed in the RO task list.

From the PVNGS SRO-Only Master Task List:

MASTER TASK LIST

Task list for OPTRNG at 2020/01/29: (189524) Senior Reactor Operator
All Tasks

Task#	Task	Selected for Training	Recurring	How Often	Training Setting
L392177	Transfer command and control of the Emergency Coordinator functions	Yes	No		Classroom
L392178	Perform the duties of the Emergency Coordinator	Yes	Yes	Yearly	Initial: Classroom & Simulator Continuing: Simulator

Technical Reference:	40EP-9EO01 Standard Post Trip Actions

If the reactor fails to automatically trip, the contingency column will be performed. The sequence is to first manually trip the reactor using RTSB push buttons (B05). If that fails, open the NGN-L03/L10 load center feeder breakers from B01. If that fails, AOs will be dispatched to trip the reactor locally.

2. Determine that Reactivity Control acceptance criteria are met by the following:

   a. Check that reactor power is dropping.

a.1 Manually trip the Reactor.

a.2 **IF** the Reactor is **NOT** tripped, **THEN** open **BOTH** of the following supply breakers:

- NGN-L03B2
- NGN-L10B2

a.3 **IF** the Reactor is **NOT** tripped, **THEN** direct an operator to open the reactor trip breakers.

a.4 **IF BOTH** of the following:

- The Reactor is **NOT** tripped
- The reactor trip breakers are **NOT** accessible

THEN direct an operator to locally open **BOTH** of the following CEDM MG Set breakers:

- NGN-L03C4
- NGN-L10C4

Technical Reference:	40DP-9AP06 Standard Post Trip Actions Technical Guideline

B. Contingency Actions

If the reactivity control safety function is not met because an automatic RPS actuation was not initiated when plant conditions required a reactor trip, guidance is provided to ensure the Reactor and Main Turbine are tripped. If the reactivity control safety function is not met because any full strength CEA is stuck out, guidance is provided to ensure the Reactor is shutdown.

- Using all four manual trip pushbuttons will ensure that power is removed from the undervoltage and shunt trip coils associated with the reactor trip switchgear breakers.
- De-energizing Load Centers L03 and L10 remotely from the control room provides an alternate method to remove power from the CEDMs.
- An operator may also be dispatched to open the reactor trip circuit breakers locally. This would be a last choice because the reactor trip switchgear is located outside the control room.

Technical Reference:	E-plan EAL Hot Chart

SU6.1 (UE) would be declared. Automatic reactor trip failed. Reactor was able to be shutdown from B01. Alert would be applicable if the reactor had to be tripped from outside the MCR.

ALERT	UNUSUAL EVENT
Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor	**Automatic or manual trip fails to shut down the reactor**
SA6.1 1 An automatic or manual trip fails to shut down the reactor as indicated by reactor power > 5% **AND** Manual trip actions taken at the reactor control consoles (B05 or B01) are **not** successful in shutting down the reactor as indicated by reactor power > 5% (Note 8)	**SU6.1** 1 An automatic trip did **not** shut down the reactor as indicated by reactor power > 5% after **any** RPS setpoint is exceeded **AND** A subsequent automatic trip or manual trip action taken at the reactor control consoles (B05 or B01) is successful in shutting down the reactor as indicated by reactor power ≤ 5% (Note 8) **SU6.2** 1 A manual trip did **not** shut down the reactor as indicated by reactor power > 5% after **any** manual trip action was initiated **AND** A subsequent automatic trip or manual trip action taken at the reactor control consoles (B05 or B01) is successful in shutting down the reactor as indicated by reactor power ≤ 5% (Note 8)

Table S-4 Communications Methods

System	Onsite	ORO	NRC

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Inadequate Core Cooling: Ability to recognize system parameters that are entry-level conditions for Technical Specifications	**Tier**			**1**
	Group			**2**
	K/A	**074 G 2.2.42**		
	IR			**4.6**

Question 83

Given the following conditions:

- Unit 3 is in MODE 4, cooling down for a refueling outage
- Train ‘A’ LPSI Pump is being used for Shutdown Cooling
- Train ‘B’ LPSI Pump is OOS for emergent corrective maintenance
- All RCPs are stopped but capable of being started

Subsequently:

- The ‘A’ LPSI Pump tripped on 86 lockout
- Work Control reports that neither LPSI Pump will be available for at least 4 hours

In order to restore compliance with LCO 3.4.6, RCS Loops – MODE 4, the crew must ensure BOTH SGs have at least a MINIMUM level of 25% ___(1)___ and start a MINIMUM of ___(2)___ RCP(s).

A. (1) wide range
   (2) one

B. (1) wide range
   (2) two

C. (1) narrow range
   (2) one

D. (1) narrow range
   (2) two

Proposed Answer:	A

Explanations: **Tech Spec evaluation during an accident (ICC) is not operationally relevant, therefore we chose to match the K/A by asking a question in which an LCO for RCS Loops is not met (another version of inadequate core cooling – in this case, inadequate RHR to meet the LCO)**	
A.	Correct.
B.	First part is correct. Second part is plausible since one loop must be in operation and each loop contains two RCPs, however only one RCP must be running for the loop to be considered "operating".
C.	First part is plausible since the top of the SG U-tubes are ~ 23.5%, therefore the statement in the SR which says, "verify secondary side water level in required SG(s) is ≥ 25%" could be interpreted as 25% NR. 25% referring to WR level is information only found in the TS bases. Second part is correct.
D.	First part is plausible since the top of the SG U-tubes are ~ 23.5%, therefore the statement in the SR which says, "verify secondary side water level in required SG(s) is ≥ 25%" could be interpreted as 25% NR. 25% referring to WR level is information only found in the TS bases. Second part is plausible since one loop must be in operation and each loop contains two RCPs, however only one RCP must be running for the loop to be considered "operating".

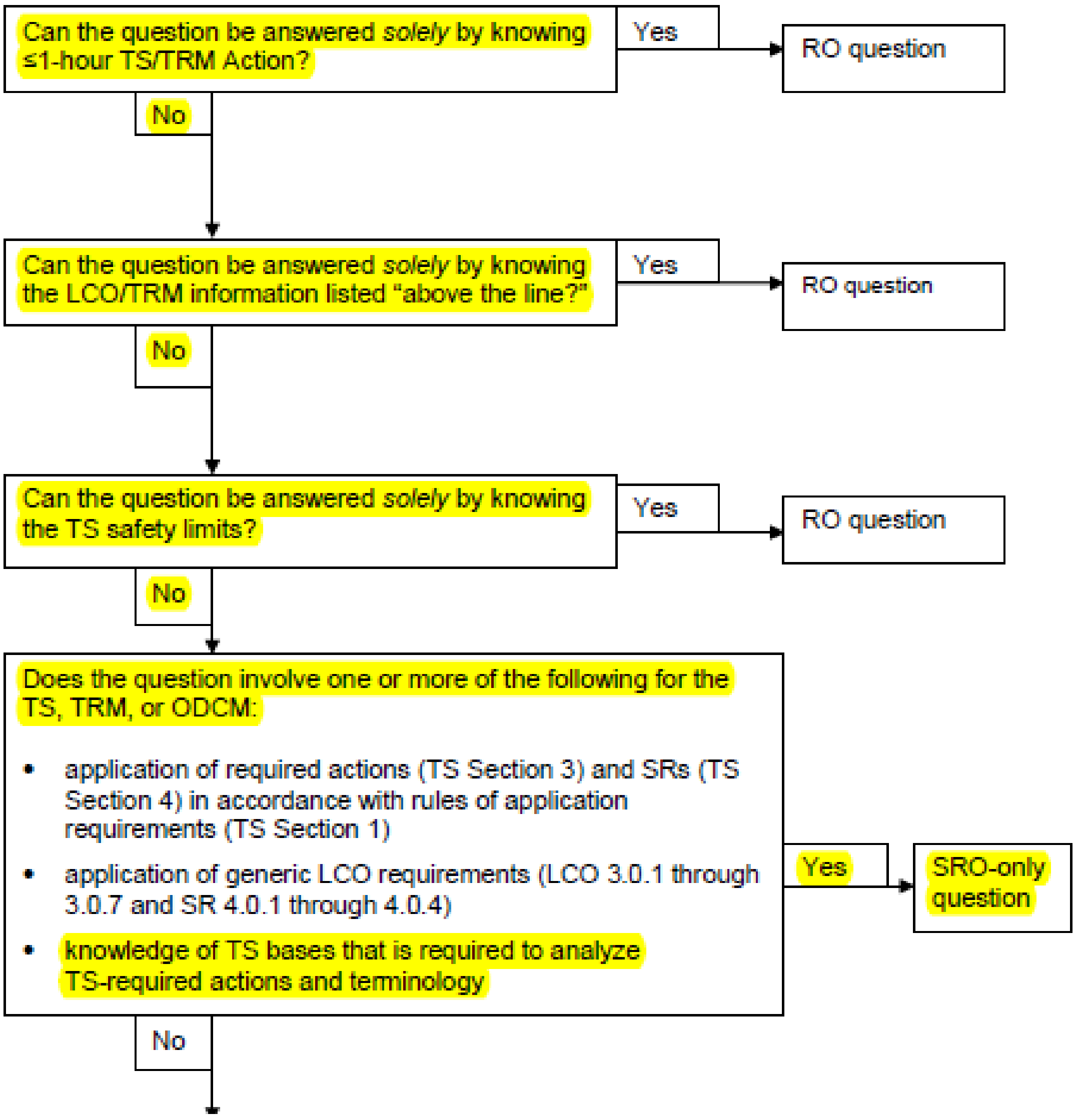
Question Source:		New	
	x	Bank – modified from 2016 Q78, but not to the point where it would be considered a "modified" question	
		Modified	
	X	Previous NRC Exam	2016 NRC Q78

Cognitive Level:	x	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	2	
Reference Provided:	N	
Learning Objective:	Identify the basis of Technical Specification LCOs and TLCOs for section 3.4 in accordance with Tech Spec 3.4 basis.	

Technical Reference:	NUREG 1021 SRO-Only Guidance

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Technical Reference:	Original Question – 2016 NRC Q78

Proposed Question: SRO 78

Given the following conditions:

- Unit 3 is in MODE 4, cooling down for a refueling outage.
- Train 'A' and Train 'B' LPSI Pumps are being used for Shutdown Cooling.
- All RCPs are secured.

Subsequently:

- The 'B' LPSI Pump indicates no flow and lower than normal amps.

In order for the unit to be in compliance with LCO 3.4.6, RCS Loops – MODE 4, at least one RCP must be OPERABLE _____(1)_____ and the associated SG must be at a MINIMUM level of 25% _____(2)_____ .

A. 1. ONLY
   2. wide range

B. 1. ONLY
   2. narrow range

C. 1. AND running
   2. wide range

D. 1. AND running
   2. narrow range

Technical Reference:	Technical Specifications

3.4.6 RCS Loops - MODE 4

LCO 3.4.6 Two loops or trains consisting of any combination of RCS loops and shutdown cooling (SDC) trains shall be OPERABLE and at least one loop or train shall be in operation.

-----NOTES-----

1. All reactor coolant pumps (RCPs) and SDC pumps may be de-energized for ≤ 1 hour per 8 hour period, provided:

   a. No operations are permitted that would cause reduction of the RCS boron concentration; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. No RCP shall be started with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR unless the secondary side water temperature in each Steam Generator (SG) is < 100°F above each of the RCS cold leg temperatures.

3. No more than 2 RCPs may be in operation with RCS cold leg temperature ≤ 200°F. No more than 3 RCPs may be in operation with RCS cold leg temperature > 200°F but ≤ 500°F.

APPLICABILITY: MODE 4.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify one RCS loop or SDC train is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.6.2	Verify secondary side water level in required SG(s) is ≥ 25%.	In accordance with the Surveillance Frequency Control Program

Technical Reference:	Technical Specifications Bases (LCO 3.4.6)

LCO	The purpose of this LCO is to require that at least two loops or trains, RCS or SDC, be OPERABLE in MODE 4 and one of these loops or trains be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS and SDC System loops. Any one loop or train in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop or train is required to be OPERABLE to provide redundancy for heat removal.
LCO (continued)	Note 2 requires secondary side water temperature in each SG is < 100¡F above each of the RCS cold leg temperatures before an RCP may be started with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR. Satisfying the above condition will preclude a large pressure surge in the RCS when the RCP is started. Note 3 restricts RCP operation to no more than 2 RCPs with RCS cold leg temperature ≤ 200°F, and no more than 3 RCPs with RCS cold leg temperature >200°F but ≤ 500°F. Satisfying these conditions will maintain the analysis assumptions of the flow induced pressure correction factors due to RCP operation (Ref. 1) An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE and has the minimum water level specified in SR 3.4.6.2.
SURVEILLANCE REQUIREMENTS (continued)	SR 3.4.6.2 This SR requires verification of secondary side water level in the required SG(s) ≥ 25% wide range. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the reactor coolant. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Technical Reference:	Technical Specifications Bases (LCO 3.4.4)

LCO 3.4.4 does require both RCPs in each loop to be operable. This is not the case for Mode 4, which only requires one RCP in the loop.

LCO

The purpose of this LCO is to require adequate forced flow for core heat removal. Flow is represented by having both RCS loops with both RCPs in each loop in operation for removal of heat by the two SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power.

Each OPERABLE loop consists of two RCPs providing forced flow for heat transport to an SG that is OPERABLE. SG, and hence RCS loop, OPERABILITY with regard to SG water level is ensured by the Reactor Protective System (RPS) in MODES 1 and 2.

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: RCS Overcooling-Pressurized Thermal Shock: Ability to determine and interpret the following as they apply to the (RCS Overcooling): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments	**Tier**		[redacted]	**1**
	Group		[redacted]	**2**
	K/A	**CE A11 AA2.2**		
	IR		[redacted]	**3.4**

Question 84

Given the following conditions:

- Unit 1 was tripped due to an ESD inside Containment
- Current plant conditions are as follows:
  - SIAS/CIAS/MSIS/CSAS have all actuated
  - All RCPs have been stopped
  - RCS Tcold is 420°F
  - REPCET is 515°F
  - RCS Pressure is 1850 psia
  - Indicated HPSI flow is 0 gpm
  - Containment temperature is 210°F
  - Containment pressure is 15 psig
  - Containment Spray flow on Train 'A' is 4800 gpm
  - Containment Spray flow on Train 'B' is 0 gpm
  - Hydrogen Analyzers have NOT yet been placed in service

Per 40EP-9EO05, ESD, the RCS Pressure Control Safety Function is ___(1)___ and the Containment Temperature and Pressure Control Safety Function is ___(2)___ .

A. (1) MET
(2) MET

B. (1) MET
(2) NOT met

C. (1) NOT met
(2) MET

D. (1) NOT met
(2) NOT met

Proposed Answer:	A

Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible since Condition 1 is not met due to Containment pressure being > 8.5 psig, and they may believe that since H2 Analyzers are not in service that the SF is not met since hydrogen cannot be verified – which would make both Condition 1 and 2 not met, however hydrogen is not required to be considered until the analyzers are in service, and even though one train of CS is 0 gpm, only one train is required to be above the minimum flow of 4350 gpm.
C.	First part is plausible since the P/T limits are exceeded based on RCS subcooling being > 200°F (if the applicant uses Tcold to determine subcooling) and SI flow being 0 gpm, however at 1850 psia in the RCS, no SI flow is required, making the RCS pressure control safety function met. Second part is correct.
D.	First part is plausible since the P/T limits are exceeded based on RCS subcooling being > 200°F (if the applicant uses Tcold to determine subcooling) and SI flow being 0 gpm, however at 1850 psia in the RCS, no SI flow is required, making the RCS pressure control safety function met. Second part is plausible since Condition 1 is not met due to Containment pressure being > 8.5 psig, and they may believe that since H2 Analyzers are not in service that the SF is not met since hydrogen cannot be verified – which would make both Condition 1 and 2 not met, however hydrogen is not required to be considered until the analyzers are in service, and even though one train of CS is 0 gpm, only one train is required to be above the minimum flow of 4350 gpm.

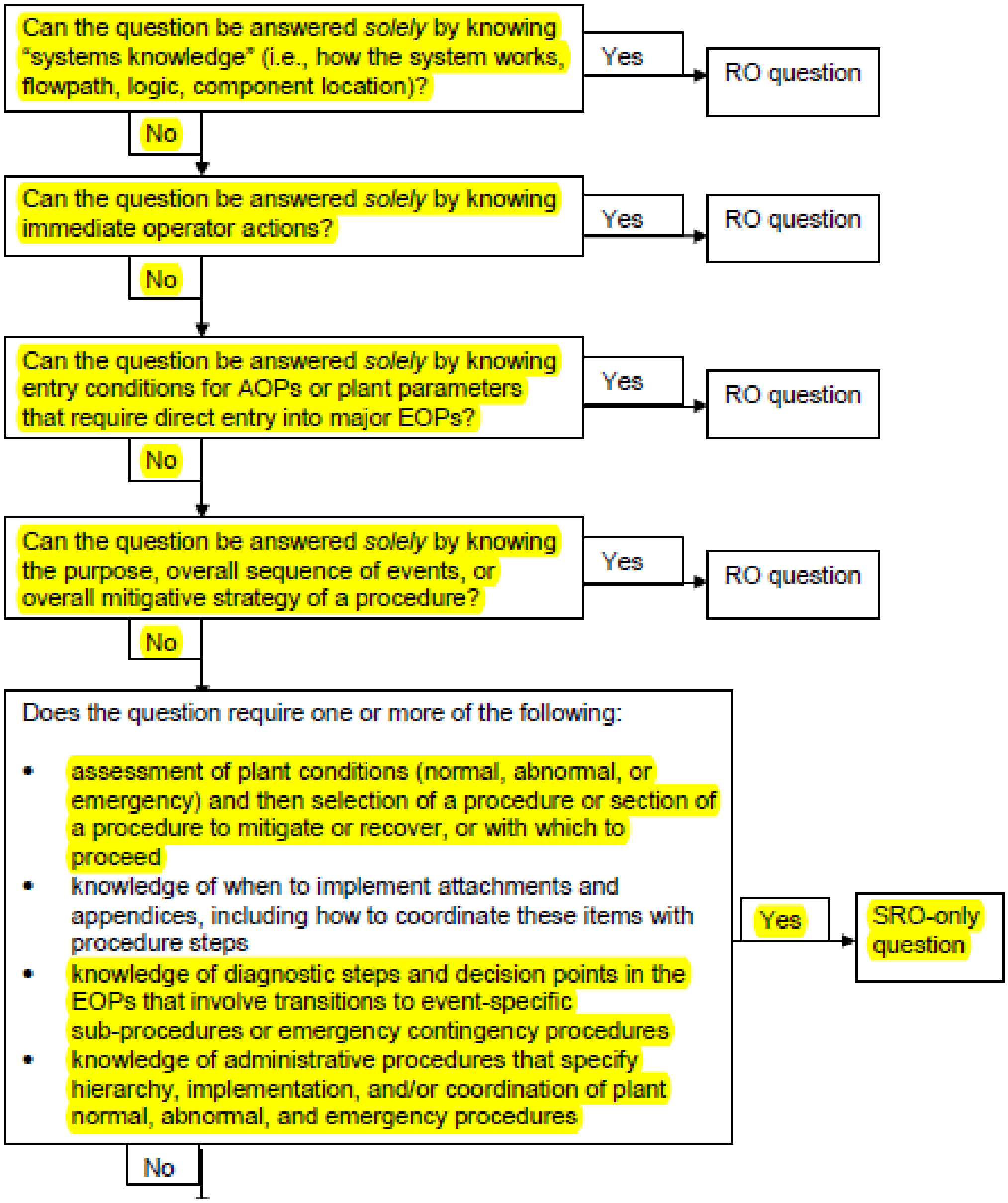
Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	5	
Reference Provided:	N	
Learning Objective:	Given conditions of an ESD, analyze RCS Pressure Control to determine if the SFSC acceptance criteria are satisfied per 40EP-9EO05.	

Technical Reference:	NUREG 1021 SRO-Only Guidance

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



Technical Reference:	40EP-9EO05 Excess Steam Demand

4. RCS Pressure Control

---------- **NOTE** ----------

Meeting the provisions of Condition 1 or Condition 2 will satisfy the RCS Pressure Control Safety Function.

ACCEPTANCE CRITERIA: **CRITERIA SATISFIED**

Condition 1

a. Pressurizer pressure is being maintained within the P/T limits. REFER TO Appendix 2, Figures. ☐ ☐ ☐ ☐

Condition 2

a. Safety Injection flow is adequate. REFER TO Appendix 2, Figures. ☐ ☐ ☐ ☐

8. Containment Temperature and Pressure Control

---------- **NOTE** ----------

Meeting the provisions of Condition 1 or Condition 2 will satisfy the Containment Temperature and Pressure Control Safety Function.

---------- **NOTE** ----------

Hydrogen criterion may be omitted until hydrogen monitor is in service.

ACCEPTANCE CRITERIA: **CRITERIA SATISFIED**

Condition 1

a. Containment temperature is less than 235°F. ☐ ☐ ☐ ☐

b. Containment pressure is less than 8.5 psig. ☐ ☐ ☐ ☐

c. Hydrogen concentration is less than 1.1%. ☐ ☐ ☐ ☐

Condition 2

a. At least one Containment Spray header flow is greater than 4350 gpm. ☐ ☐ ☐ ☐

b. Containment pressure is less than 55 psig. ☐ ☐ ☐ ☐

c. Hydrogen concentration is less than 4.9%. ☐ ☐ ☐ ☐

Technical Reference:	40DP-9AP10 Excess Steam Demand Technical Guideline

4.6.4 SFSC #4 - RCS Pressure Control

A. The intent of the Pressure Control Safety Function is to ensure that adequate subcooling exists.

Condition 1

RCS pressure control is satisfactory if the RCS can be maintained within the post-accident P/T limits. Maintaining the RCS within the limits maintains subcooling necessary for single phase natural circulation flow and minimizes the possibility of PTS.

Condition 2

If pressurizer pressure cannot be maintained within the post-accident P/T limits, then pressure control is satisfied by ensuring adequate SI flow. Once SI flow has been throttled or stopped or a RAS has occurred, the SI delivery Curves are no longer applicable.

4.6.8 SFSC #8 - Containment Temperature and Pressure Control

A. The intent of the Containment Temperature and Pressure Control Safety Function is to check that the containment environment is maintained within design temperature and pressure limits.

The containment is also monitored for hydrogen once the hydrogen monitors are in service. Hydrogen build up to levels greater than the minimum level of detection is not expected for accidents that are within design basis. However, in the event of a beyond design basis accident with fuel damage, there may be a buildup of hydrogen in containment. Hydrogen monitoring is initiated in the EOPs to support implementation of Severe Accident Management Guideline (SAMG) strategies in the event of a beyond design basis accident, should they ultimately be needed to respond to the event.

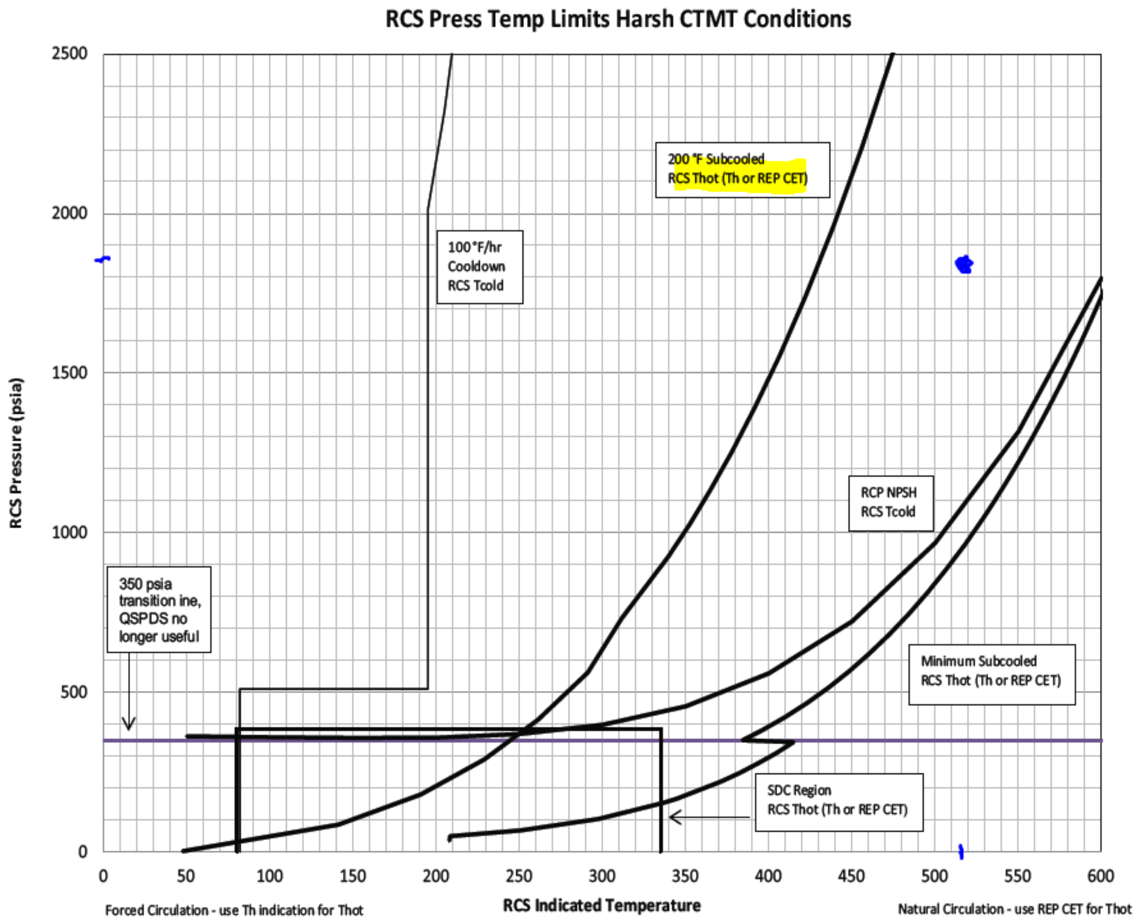
Condition 1 uses 1.1% for the hydrogen criteria, while Condition 2 uses 4.9%. While it is possible hydrogen may be present in the containment following an accident, it is not expected. Specifying a lower (less restrictive) limit in Condition 1 and the higher (more restrictive) limit in Condition 2 is consistent with industry emergency procedure development past practices. If hydrogen is greater than 1.1% but below 4.9%, it is acceptable to stay in 40EP-9EO05. However, if hydrogen concentration exceeds 4.9%, 40EP-9EO05 should be exited and 40EP-9EO09, Functional Recovery, implemented. In addition, the TSC should be notified to provide further guidance

Condition 2

Containment temperature and pressure may exceed the above limits during inside containment ESD events. If this happens, containment spray should be operating to minimize the temperature and pressure inside containment. At least one containment spray header delivering greater than the minimum acceptable flow will remove 100% of the design basis heat load.

Hydrogen concentration is less than the lower flammability concentration. The intent of this application is to prevent a containment-wide hydrogen burn to avoid exceeding the containment pressure/temperature assumed in the safety analysis and minimize damage to safety-related equipment located in containment. Excessive hydrogen concentrations may be possible following a LOCA if there is fuel damage, but mitigating actions are outside the realm of the EOPs. Placing the hydrogen monitors in service ensures they will be available to the TSC for use in the event of entry into the Severe Accident Management Guidelines. A note is provided to inform the operator that hydrogen concentration acceptance criteria may be omitted until the hydrogen monitor is in service. This is because it takes time to place the monitors in service and obtain reliable indication.

Technical Reference:	40EP-9EO10-002 Appendix 2: Figures



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Excess RCS Leakage: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits	Tier			1
	Group			2
	K/A	CE A16 G 2.2.25		
	IR			4.2

Question 85

Given the following conditions:

- Unit 1 is in MODE 1
- **At time = 0200 on 5/5/2021:** The CRS entered 40AO-9ZZ02, Excessive RCS Leakrate, due to indications of an RCS leak inside Containment
    - Containment sump monitors indicate a 7 gpm rise in sump levels
    - The water inventory balance indicates an RCS leak rate of 22 gpm
- **At time = 0400 on 5/5/2021:** A Containment entry was made and the source of the leak was determined to be from a failed RCS pipe weld

Assuming the leak cannot be isolated, in order to comply with LCO 3.4.14, RCS Operational LEAKAGE, the LATEST time Unit 1 can enter MODE 5 is _____ on 5/6/2021.

A. 1400

B. 1600

C. 1800

D. 2000

Proposed Answer:	B

Explanations:	
A.	Plausible if thought that since the leak started at 0200 on 5/5 that the action for MODE 5 in 36 hours would be effective at the time the leak started, however the MODE 5 in 36 hours doesn't start until the leakage is confirmed to be pressure boundary leakage, which didn't happen until 0400.
B.	Correct.
C.	Plausible if though that the full 4 hours allotted in 3.4.14 condition A applies prior to entering condition B, however condition B is applicable as soon as the leakage is confirmed to be pressure boundary leakage.
D.	Plausible if thought that since the leak started at 0200, and it ended up being pressure boundary leakage that the required actions of condition B applied as of 0200, and if thought that the required action was to be in MODE 3 in 6 hours and then had an additional 36 hours to be in MODE 5.

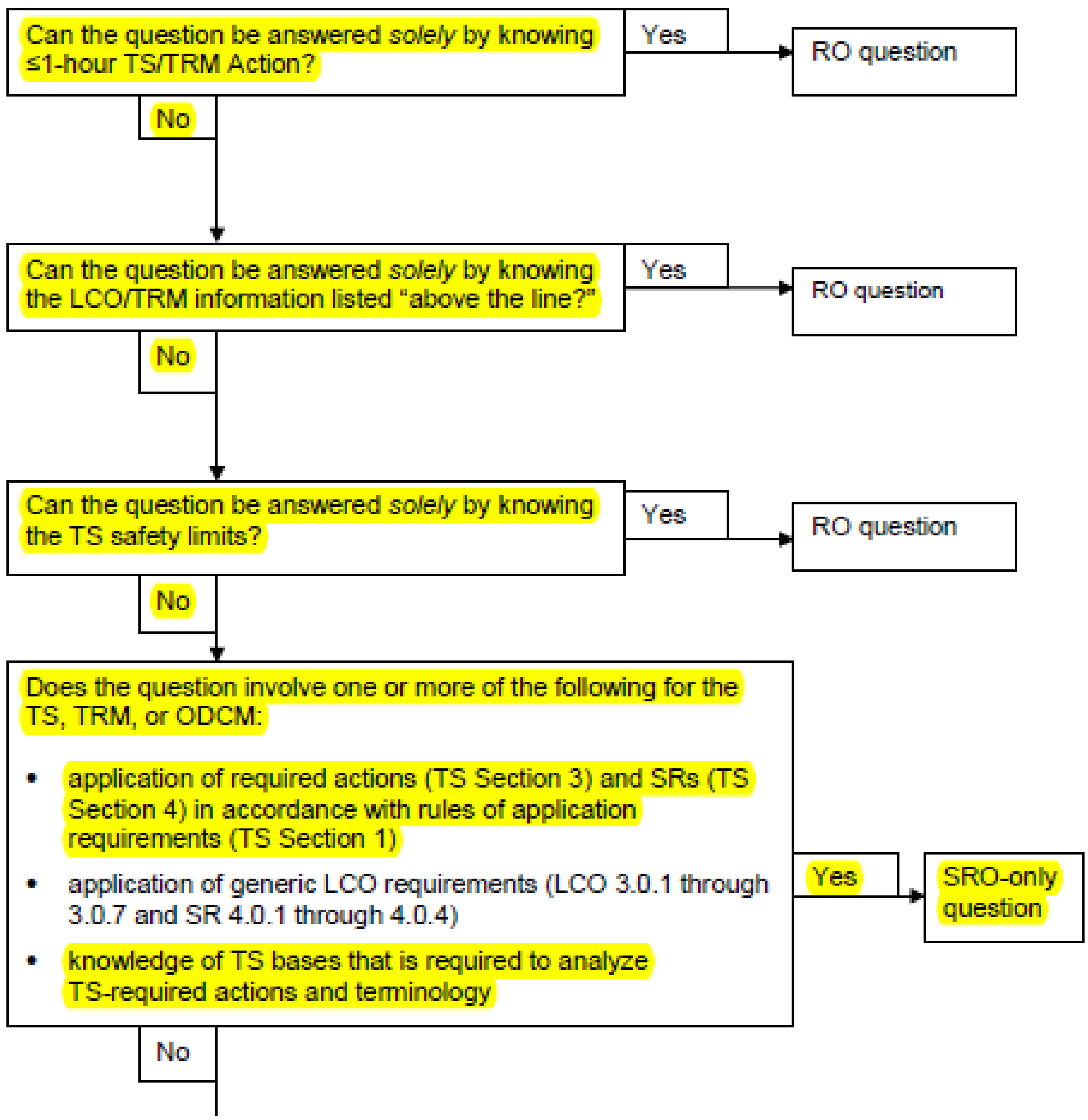
Question Source:		**New**	
	X	**Bank**	
		Modified	
		Previous NRC Exam	

Cognitive Level:		**Memory or Fundamental Knowledge**
	X	**Comprehension or Analysis**

Level of Difficultly:	**4**	
10CFR55.43:	**2**	
Reference Provided:	**N**	
Learning Objective:	**Given plant conditions and Technical Specification action statements that are greater than one hour, apply the action statements that are greater than one hour for LCOs and TLCOs of 3.4 in accordance with Tech Spec 3.4.**	

Technical Reference:	NUREG 1021 SRO-Only Guidance

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Technical Reference:	Technical Specifications

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Operational LEAKAGE

LCO 3.4.14 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

Technical Reference:	Tech Spec Bases

The leakage, although pressure boundary, cannot be classified as pressure boundary until visually observed, so the required actions for pressure boundary leakage do not go into effect until 0400, when the leakage is observed to be from a pipe weld.

BASES

ACTIONS

B.1 and B.2 (continued)

4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and to MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Component Cooling Water: Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High/low CCW temperature	**Tier**			**2**
	Group			**1**
	K/A	**008 A2.03**		
	IR			**3.2**

Question 86

Given the following conditions:

- Unit 2 is operating at 100% power, MOC

Subsequently:

- A failure in CHN-TIC-223, Letdown HX Outlet Temp Control, has resulted in MAXIMUM NC flow through the HX

(1) With NO operator action, this failure will cause Reactor power to _____ .

(2) Per the PVNGS Event Reporting Manual, a 24-hour notification to the NRC is required AS SOON AS power _____ .

A. (1) rise
(2) exceeds 102%

B. (1) rise
(2) exceeds 105%

C. (1) lower
(2) drops below 98%

D. (1) lower
(2) drops below 95%

Proposed Answer:	A

Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible since a 5% unplanned power change does trigger a notification per the PV Event Reporting Manual, however in this case the event is first reportable after a 2% change.
C.	First part is plausible since a change in letdown temperature will change the affinity for boron absorption in the IX, however lowering letdown temp will result in a dilution, causing Reactor power to rise. Second part is plausible since a 2% change is what is reportable in this case, however it is a 2% rise in power, not lowering.
D.	First part is plausible since a change in letdown temperature will change the affinity for boron absorption in the IX, however lowering letdown temp will result in a dilution, causing Reactor power to rise. Second part is plausible since a 5% unplanned power change does trigger a notification per the PV Event Reporting Manual, however in this case, the event is first reportable when the power change exceeds 2%.

Question Source:	x	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	x	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	1	
Reference Provided:	N	
Learning Objective:	As an SRO, describe the reporting requirements associated with exceeding licensed thermal power output, per the Event Reporting Manual.	

Technical Reference:	NUREG 1021 SRO-Only Guidance

II. Examples of Additional Knowledge and Abilities as They Pertain to an SRO License and the 10 CFR 55.43(b) Topics [ES-401, Section D.1.c]

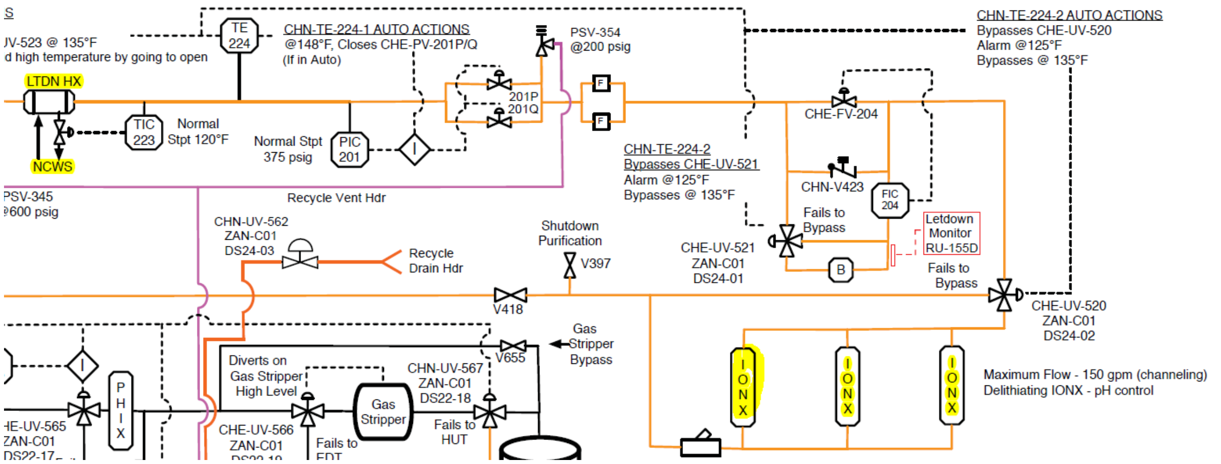
A. Conditions and Limitations in the Facility License [10 CFR 55.43(b)(1)]

Examples of SRO exam items for this topic include the following:

- reporting requirements when the maximum licensed thermal power output is exceeded
- administration of fire protection program requirements, such as compensatory actions associated with inoperable sprinkler systems and fire doors
- required actions necessary when a facility does not meet the administrative controls listed in Technical Specifications (TS), Section 5 or 6, depending on the facility (e.g., shift staffing requirements)
- National Pollutant Discharge Elimination System requirements, if applicable
- processes for TS and final safety analysis report changes

Technical Reference:	PVGS Operator Information Manual

Max NCW flow through the CVCS letdown HX will lower the temperature of the letdown water. This will cool the resin in the letdown demins (IXs). Cooler temperature cause the IX resins to have a higher affinity for boron (abosorbs more boron). This results in a dilution of the RCS water, cause reactor power to rise.



The boron affinity of a resin bed is affected by the temperature of the coolant passed through the bed. At lower temperatures, the borate ion bonding to the exchange site contains three boron atoms. At higher temperatures, the borate ion contains only one boron atom. The result of this characteristic is that at lower temperatures the resins are more efficient at removing boron from the coolant than at higher temperatures. A saturated resin bed will actually release boron as temperature is increased.

If the coolant going through the ion exchanger is at a low temperature, the affinity of the ion exchange resin bead exchange site is high. A borate ion $(BO_3)^{-3}$ in a low temperature solution passing through the ion exchanger will be adsorbed (exchanged) and removed from the coolant. This reduces the number of boron atoms in coolant.

If the coolant going through the ion exchanger is at a high temperature, the affinity of the ion exchange resin bead exchange site is low. If an ion higher on the affinity list than borate $(BO_3)^{-3}$ passes through the ion exchanger, that ion will be adsorbed by the resin exchange site and a borate ion $(BO_3)^{-3}$ will be released and added to the coolant. This increases the number of boron atoms in coolant.

Technical Reference:	40OP-9CH01 CVCS Normal Operations

6.2 Adjusting CHN-TIC-223, Letdown Heat Exchanger Outlet

NOTE

___ • Lowering the letdown temperature raises the Nuclear Cooling Water flow rate on the shell side of the CHN-E02, Letdown Heat Exchanger (LDHX).

- Normal design setpoint for CHN-TIC-223 is 120°F.
- The CHN-TIC-223 setpoint should be maintained greater than or equal to 112°F whenever conditions allow.
- Lowering the letdown temperature causes the resin in the CVCS Ion Exchangers to have a slightly more apparent capacity for boron, which has the delayed effect similar in magnitude to a small dilution.

Technical Reference:	Event Reporting Manual

Operation >102% is reportable due to being an unanalyzed condition (8 hour report)

Maximum Power Level

The average power level over any 8-hour shift should not exceed the "full steady-state licensed power level" (and similarly worded terms). The exact 8-hour periods defined as "shifts" are up to the plant, but should not be varied from day to day (the easiest definition is a normal shift manned by a particular "crew"). It is permissible to briefly exceed the "full, steady-state licensed power level" by as much as 2% for as long as 15 minutes. In no case should 102% power be exceeded, but lesser power "excursions" for longer periods should be allowed, with the above as guidance. For example, 1% excess for 30 minutes and 1/2% for 1 hour should be allowed. There are no limits on the number of times these "excursions" may occur, or the time interval that must separate such "excursions." The above requirement regarding the 8-hour average power will prevent abuse of this allowance. Operation outside these limits would constitute a reportable violation. [Ref: NRC Inspection Procedure (IP) 61706, *Core Thermal Power Evaluation,* paragraph 3.01(d)]

Maximum Power Level

Some stations do not have a license power limit stated explicitly as a license condition. Although operation above that power level not specifically identified as a reportable event in their license, exceeding the maximum authorized power level is a significant violation of the operating license which should be reported to the NRC within 24 hours followed by an LER within 30 or 60 days.

T.S. Violations

T.S. violations should be reported in accordance with 10 CFR 50.72 and 10 CFR 50.73.

Palo Verde Interpretations

The Palo Verde Facility Operating Licenses were amended (Amendment 158) in March 2006 to remove the requirement in Section 2.F (2.G in Unit 3) to report any violations of the requirements contained in Sections 2.C of the License.

Some violations may still be reportable under some other reporting requirement. For example, operation above 102% of rated thermal power is an unanalyzed condition. On the other hand, operation between 100% and 102%, while still a violation of the license, is not reportable. This, as any other violation of a requirement must be identified and entered into the corrective action program.

373	Rated thermal power level exceeded (>102% power, Violation of a License Condition)	N/A	N/A	XREF: NRRH-070 for unanalyzed condition. Link to detailed discussion

ID	EVENT DESCRIPTION	ENS	WRITTEN	SPECIFIC NOTIFICATION
070	The occurrence of any event or condition that results in the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety.	08 hr.		Follow up call to NRC may be required: NRRH-002 NRRH-005 NRRH-006 NRRH-007 NRRH-008 Source Docs: § 50.72(b)(3)(ii)(B) NUREG-1022, Section 3.2.4 Note: This is a very subjective reporting requirement. Significant additional information is in this link to detailed discussion

Technical Reference:	Event Reporting Manual

A 5% power reduction also requires internal and external notifications.

INTERNAL OPERATIONS NOTIFICATION REQUIREMENTS

Palo Verde Interpretations

The Unit Shift Manager (USM) is responsible for initiating appropriate notification to applicable Palo Verde Management in order that the following Operations Notification Requirements are met. These same levels of criteria, although no other regulatory reporting requirement may be met, normally warrant a "courtesy" notification to the NRC Resident Inspector. Following notification initiated by the USM or the Operations Department Leaders, the Operations Director, or his designee, is responsible to ensure that plant status and events are communicated to the Sr. Vice President, Nuclear, or designee, in a timely and accurate manner as specified below:

The following events are to be communicated to the Sr. Vice President, Nuclear IMMEDIATELY:

- Unplanned load reduction greater than or equal to 5%
- Unplanned turbine or reactor trip

866	UNPLANNED Power Reduction greater than 5 percent for an actual or anticipated period greater than 10 hours.	*		*Notify Communications & Participant Services

Technical Reference:	ODP-33 Operations Nuclear Regulatory Commission Communications

NRC resident is also required to be notified for a > 5% power change.

II. OVERVIEW

The Unit Shift Manager (SM) or Unit Operations Manager (UOM) is responsible to initiate NRC notification to Regulatory Affairs, the NRC Senior Resident Inspector or designee as specified below:

A. Unplanned load changes or plant transients

- Equipment issues resulting in unplanned load changes greater than 10% or unplanned Power Reduction greater than 5 percent for an actual or anticipated period greater than 10 hours.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Containment Cooling: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures	**Tier**			**2**
	Group			**1**
	K/A	**022 G 2.4.4**		
	IR			**4.7**

Question 87

Given the following conditions:

- Unit 3 was tripped from 100% power due to lowering Pressurizer level and pressure
- SIAS and CIAS were manually actuated following the Reactor trip

Assuming the CRS has transitioned to an optimal EOP, in which of the following situations, INDIVIDUALLY, should the CRS direct the crew to perform Appendix 17, Restoration of Containment Cooling?

	Containment Pressure	**Containment Temperature**	**Containment Level**
Situation 1	22 psig - lowering	210°F - lowering	15 feet - rising
Situation 2	5 psig - rising	150°F - rising	8 feet - rising
Situation 3	0.8 psig - rising	120°F - rising	Not on scale

A. Situation 1 ONLY

B. Situation 1 and 2 ONLY

C. Situation 2 and 3 ONLY

D. Situation 3 ONLY

Proposed Answer:	D

Explanations:	
A.	Plausible since situation 1 is the only one in which containment pressure is > the CSAS setpoint and temperature is above the threshold for harsh containment conditions, however while temperature and pressure are used for determining the status of the containment integrity safety function, only containment level is used when determining if containment cooling should be restored.
B.	Plausible since situations 1 and 2 are both above the containment pressure SIAS and CIAS setpoints, and SIAS does stop normal containment cooling due to the SIAS load shed scheme, however only containment level is used when determining if containment cooling should be restored.
C.	Plausible that situations 2 and 3 would warrant the restoration of containment cooling since they both have elevated containment temperature and pressure following a SIAS but do not have a CSAS actuation, however condition 2 would not warrant the restoration of containment cooling since level is indicated inside containment.
D.	Correct. Although this set of conditions are the least severe, following a SIAS actuation, the sole determiner for restoring containment cooling is whether or not containment level is on scale or not.

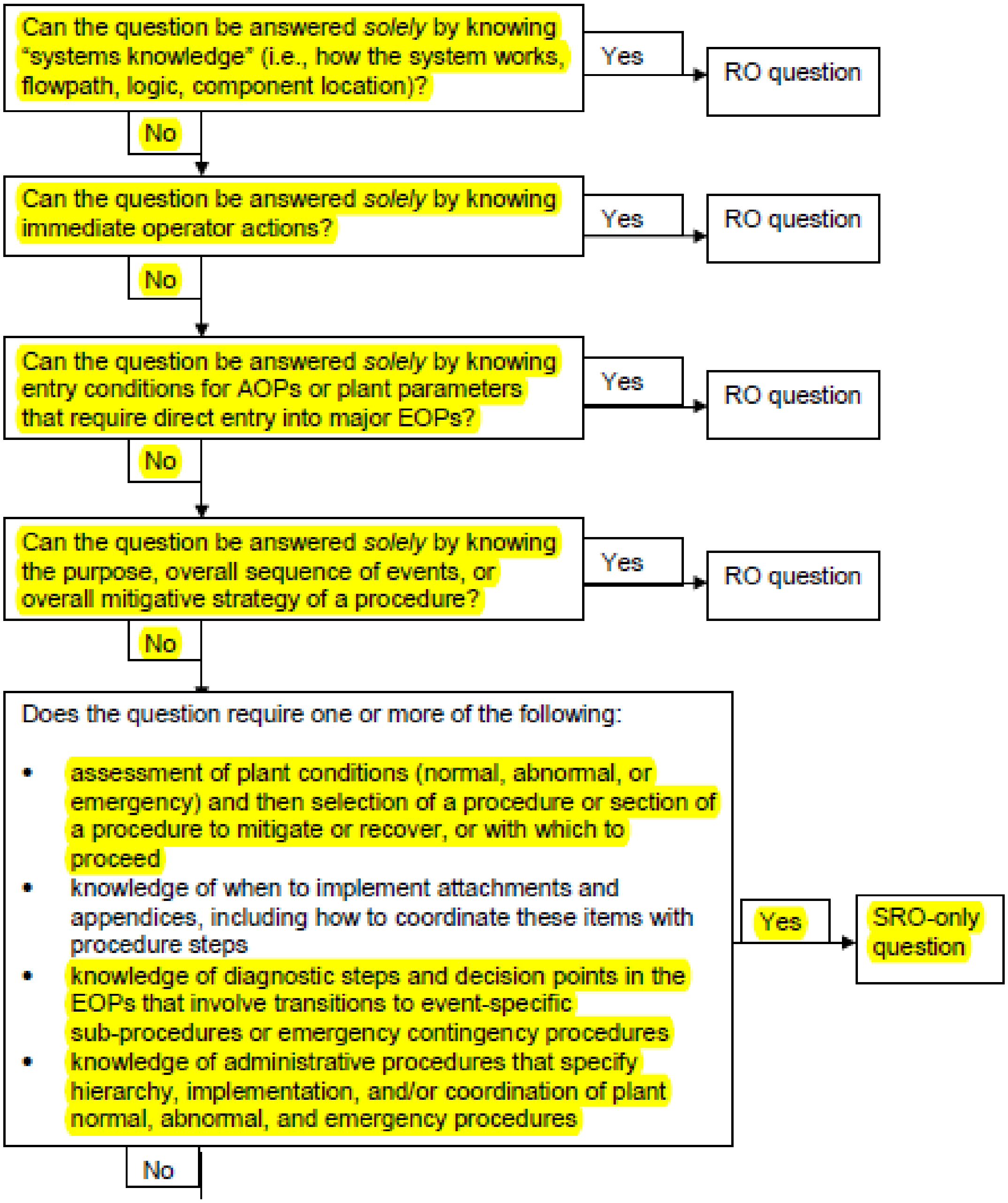
Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	**X**	**Memory or Fundamental Knowledge**
		Comprehension or Analysis

Level of Difficultly:	**3**	
10CFR55.43:	**5**	
Reference Provided:	**N**	
Learning Objective:	**Given conditions of a LOCA, describe why it is beneficial to restore normal containment cooling following a SIAS per 40EP-9EO03.**	

Technical Reference:	NUREG 1021 SRO-Only Guidance

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Technical Reference:	40EP-9EO03 Loss of Coolant Accident

Normal containment cooling is restored if there is no containment sump level indicated.

✱ 20. **IF** SIAS has actuated,
THEN perform the following:

a. Energize the SIAS Load Shed Panels. REFER TO Appendix 21, List of SIAS Load Shed Panels.

b. **IF** containment level is **NOT** indicated,
THEN PERFORM Appendix 17, Restoration of Containment Cooling.

Technical Reference:	40DP-9AP08 Loss of Coolant Accident Technical Guideline

4.5.20 Step 20 - If SIAS, Restore Systems

A. Re-energizing SIAS load shed panels is required to ensure the operability of the non-safety auxiliary feed pump, essential lighting, Containment cooling and other needed loads. Essential lighting and other needed loads can be safely restored in a controlled manner after a SIAS.

If containment level is not indicated, then normal containment cooling should be established in order to maximize recirculation of the containment atmosphere. This recirculation will minimize the possibility of local accumulations of hydrogen developing. This will also help in removing heat from the containment in order to stop containment spray as soon as possible. If containment level is indicated, normal containment cooling shall not be restored. This is due to the potential for submergence and eventual failure of the inside containment isolation valve motor operators for NC and WC.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Emergency Diesel Generator: Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Load, VARS, pressure on air compressor, speed droop, frequency, voltage, fuel oil level, temperatures	Tier			2
	Group			1
	K/A	064 A2.02		
	IR			2.9

Question 88

Given the following conditions:

- Unit 3 is operating at 100% power
- **At time = 0800:**
  - ‘A’ EDG Starting Air Receiver A, DGA-X01A, was tagged out for corrective maintenance
- **At time = 0900:**
  - An AO reports a small air leak in ‘A’ EDG Starting Air Receiver B, DGA-X01B
  - Current Air Receiver B pressure is 230 psig and is lowering at a rate of 1 psig/minute

If Air Receiver B pressure continues to lower at the current rate, 40ST-9ZZ37, Inoperable Power Sources Action Statement, must FIRST be performed NO LATER THAN…

A. 1000

B. 1015

C. 1045

D. 1100

Proposed Answer:	C

Explanations:	
A.	Plausible if thought that the EDG is inoperable when the second Air Receiver drops below 230 psia, however the action for the second receiver < 230 psia is to restore one receiver within 48 hours, and the EDG is inoperable when the second receiver is < 185 psia. The assumption of 1 hour to perform SR 3.8.1.1 (which is performance of 40ST-9ZZ37) within 60 minutes would be correct.
B.	Plausible if thought that the EDG is inoperable when the second Air Receiver drops below 230 psia, however the action for the second receiver < 230 psia is to restore one receiver within 48 hours, and the EDG is inoperable when the second receiver is < 185 psia. 1015 would be correct if the assumption of the time of EDG inoperability were correct and if the 1.25 SR frequency extension were applied, however for the first performance of the SR, SR 3.0.2 is not applicable.
C.	Correct.
D.	The time of EDG inoperability is correct, however 1100 would indicate that the 1.25 SR frequency extension is able to be applied, however the first performance of a “once within xxx and once every xxx…” does not allow for the 1.25 time extension.

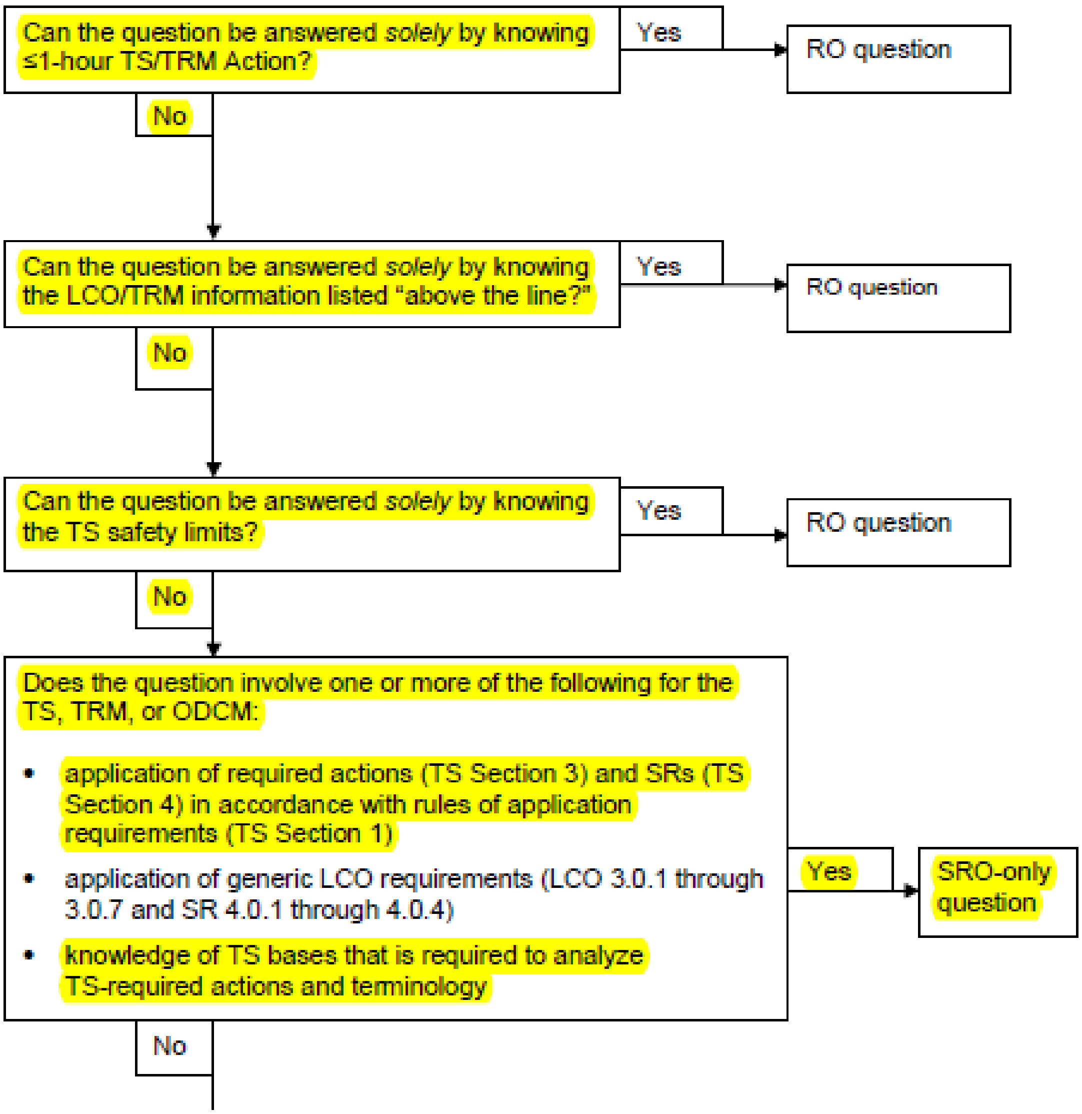
Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		**Memory or Fundamental Knowledge**
	X	**Comprehension or Analysis**

Level of Difficultly:	**4**	
10CFR55.43:	**2**	
Reference Provided:	**N**	
Learning Objective:	**Identify the bases of Technical Specification LCOs and TLCOs for section 3.8 in accordance with Tech Spec 3.8 bases.**	

Technical Reference:	**NUREG 1021 SRO-Only Guidance**

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Technical Reference:	Technical Specifications

25% extension cannot be used for the initial performance of SR 3.8.1.1. It could be used for the subsequent 8 hours performances.

SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per ..." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

Example 1.3-7 explains the completion time rules for a similar required action as LCO 3.8.1 required action B.1.

EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND B.2 Be in MODE 5.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered.

Technical Reference:	Technical Specifications (LCO 3.8.3)

When the air receiver drops to < 230 psig (0900), LCO 3.8.3 Condition E is entered.

With the pressure dropping at 1 psig/min, the air receiver drops to < 185 psig in 45 minutes (0945). When this occurs, LCO 3.8.3 Condition F is entered. Which requires the associated DG to be declared inoperable.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more DGs with new fuel oil properties not within limits.	D.1 Restore stored fuel oil properties to within limits.	30 days
E. One or more DGs with a required starting air receiver pressure < 230 psig and ≥ 185 psig.	E.1 Restore starting air receiver pressure to ≥ 230 psig.	48 hours
F. Required Action and associated Completion Time not met. OR -----------NOTE--------- Should the required starting air receiver pressure momentarily drop to <185 psig while starting the DG on one air receiver only, then entry into Condition F is not required. ----------------------------- One or more DGs with diesel fuel oil, lube oil, or starting air subsystem inoperable for reasons other than Condition A, B, C, D, or E.	F.1 Declare associated DG inoperable.	Immediately

Technical Reference:	Technical Specifications (LCO 3.8.1 Condition B))

DG is declared inoperable at 0945. LCO 3.8.1 Condition B is entered.

Required Action B.1 must first be performed within 1 hour (by 1045). The 25% surveillance extension cannot be used for the first performance.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 Restore required offsite circuit to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. One DG inoperable.	B.1 Perform SR 3.8.1.1 for the OPERABLE required offsite circuit(s).	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>	
	B.2 Declare required feature(s) supported by the inoperable DG inoperable when its redundant required feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	<u>AND</u>	

Technical Reference:	40ST-9ZZ37 Inoperable Power Sources Action Statement

40ST-9ZZ37 is performed to comply with LCO 3.8.1 required actions.

Inoperable Power Sources Action Statement	40ST-9ZZ37	Revision 6

1.0 PURPOSE AND SCOPE

1.1 Purpose

1.1.1 Provide specific guidance for complying with the action statements A through G of LCO 3.8.1 AC Sources - Operating.

1.2 Scope

1.2.1 General

A. Address ANY of the following circumstances:

- One required off site circuit inoperable.
- One diesel generator inoperable.
- Two required off site circuits inoperable.
- One required off site circuit and one diesel generator inoperable.
- Two diesel generators inoperable.
- One automatic load sequencer inoperable.

1.2.2 Technical Specification Requirements

A. LCO 3.8.1, AC Sources - Operating.

B. SR 3.8.1.1 Verification of required off site power circuit.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Process Radiation Monitoring: Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Erratic or failed power supply	Tier			2
	Group			1
	K/A	073 A2.01		
	IR			2.9

Question 89

Given the following conditions:

- A Waste Gas Decay Tank release is required
- Gaseous Radwaste Radiation Monitor, RU-12, has just failed due to a short circuit in the power supply

Which of the following describes the required action(s) in order to perform the release as planned?

In order for the release to be performed, ___(1)___ as required by ___(2)___ .

A. (1) initiate the Preplanned Alternate Sampling Program to monitor the appropriate parameter
   (2) the Offsite Does Calculation Manual

B. (1) initiate the Preplanned Alternate Sampling Program to monitor the appropriate parameter
   (2) 74RM-9EF41, Radiation Monitoring System Alarm Response

C. (1) at least two technically qualified personnel must independently verify the discharge valve lineup
   (2) the Offsite Dose Calculation Manual

D. (1) at least two technically qualified personnel must independently verify the discharge valve lineup
   (2) 74RM-9EF41, Radiation Monitoring System Alarm Response

Proposed Answer:	C

Explanations:	
A.	First part is plausible since this action is required by the ODCM for a failure of RU-145, Noble Gas Activity Monitor, however this is not required for a failure of RU-12. Second part is correct.
B.	First part is plausible since this action is required by the ODCM for a failure of RU-145, Noble Gas Activity Monitor, however this is not required for a failure of RU-12. Second part is plausible since the ARP provides contingency actions for alarming or failed RMs, however there are no requirements in the ARP for gaseous releases.
C.	Correct.
D.	First part is correct. Second part is plausible since the ARP provides contingency actions for alarming or failed RMs, however there are no requirements in the ARP for gaseous releases.

Question Source:		New	
	x	Bank	
		Modified	
	x	Previous NRC Exam	2020 NRC Q87 (slightly modified but still bank question)

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	4	
Reference Provided:	N	
Learning Objective:	Given key parameter indications and various plant conditions, predict Gaseous Radwaste System normal, abnormal, and emergency operations including design basis, flowpaths, major components, and controls/interlocks in accordance with approved system specific documentation and/or approved operating procedures.	

Technical Reference:	NUREG 1021 SRO-Only Guidance

D. Radiation Hazards That May Arise during Normal and Abnormal Situations, including Maintenance Activities and Various Contamination Conditions [10 CFR 55.43(b)(4)]

Some examples of SRO exam items for this topic include the following:

- process for gaseous/liquid release approvals (i.e., release permits)
- analysis and interpretation of radiation and activity readings as they pertain to the selection of administrative, normal, abnormal, and emergency procedures
- analysis and interpretation of coolant activity, including comparison to emergency plan criteria and/or regulatory limits

Technical Reference:	74RM-9EF41 Radiation Monitoring System Alarm Response
This alarm response is generic to all Rad Monitors. If a Rad Monitor is found non-functional, you go to 74RM-9EF43, which provides further guidance on required actions or other procedures to use.	

6.4.12 **IF** ANY of the following:

- Monitor malfunction is indicated
- Equipment failure is indicated

THEN perform the following:

A. **IF** alarm response actions can NOT correct the malfunction/failure, **THEN** initiate a Condition Report (CR).

1. Perform an assessment of monitor operability/functionality per 74RM-9EF43, Actions for Nonfunctional Radiation Monitors: Preplanned Alternate Sampling Program.

2. **IF** the monitor is INOPERABLE/NONFUNCTIONAL, **THEN** perform required ACTIONS for an inoperable/nonfunctional monitor per 74RM-9EF43, Actions for Nonfunctional Radiation Monitors: Preplanned Alternate Sampling Program.

Technical Reference:	74RM-9EF43 Actions For Non-Functional Radiation Monitors: Preplanned Alternate Sampling Program

If RU-12 is non-functional, a WGDT release may occur as long as 74RM-9EF20 requirements are met. These requires comply with the ODCM actions.

Appendix A - OPERABILITY/FUNCTIONALITY Guidelines And Actions For RMS Monitors

Table 1: Technical Specification/TRM/ODCM Channels

Monitor	Channel Description	OPERABILITY / FUNCTIONALITY Guideline #	INOPERABILITY / NON-FUNCTIONALITY Action	Applicable Mode/Reference
RU-12	Noble Gas	A,B,G,H	2, 11	During Waste Gas release ODCM Table 2-1

2 If NON-FUNCTIONAL, the contents of the tank(s) may be released to the environment provided that prior to initiating the release the following actions are initiated per 74RM-9EF20, Gaseous Radioactive Release Permits and Offsite Dose Assessment.

11 If minimum required channels have not been restored to FUNCTIONAL status within 30 days, initiate a CR to explain in the next Annual Radioactive Effluent Release Report why this NON-FUNCTIONALITY was not corrected within the specified time.

For other Rad Monitors, the required actions (preplanned alternate sampling program) are contained in 74RM-9EF43. For example, RU-145, which is also an ODCM required Rad Monitor.

RU-145(b)	Noble Gas (Low)	A,B,D,E,G,H	5,8,11(d)	Modes 1, 2, 3, & 4 or whenever irradiated fuel is in the Fuel Storage Pool ODCM Table 2-1
	Iodine Sampler	D,F,K	9,10,11	
	Particulate Sampler	D,F,K	9,10,11	
	Process Flow Rate Monitor	A,G,J	10,11	
	Sampler Flow Rate Monitor	A,G,K	9,10,11	

8 If NON-FUNCTIONAL, effluent releases via this pathway may continue provided noble gas grab samples are obtained every 8 hours, not to exceed 12 hours.

8.1 Compare sample results with monitor alarm setpoints.

8.2 Notify operations if any setpoint value is exceeded.

8.3 The associated High Range monitor is also NON-FUNCTIONAL unless the High Range monitor is operating in the Handover Disable mode and is active.

9 If NON-FUNCTIONAL, effluent releases via the effected pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in ODCM Table 3-1 within one hour after the channel has been declared non-functional.

9.1 Initiate alternate sampling per 74RM-9EF63, RU-143 Sample Operations or 74RM-9EF65, RU-145 Sample Operations. If maintenance activities on the ventilation system ducts will prevent obtaining representative samples from the sample points specified in 74RM-9EF63, RU-143 Sample Operations or 74RM-9EF65, RU-145 Sample Operations, initiate continuous sample collection locally using the methods prescribed in 75RP-9RP07, Radiological Surveys and Air Sampling.

Technical Reference:	74RM-9EF20 Gaseous Radioactive Release Permits And Offsite Dose Assessment

6.4 Waste Gas Decay Tank Release Permits

6.4.1 General Instructions

A. Determine if the Waste Gas Decay Tank (WGDT) Discharge Monitor (RU-12) is FUNCTIONAL.

1. **IF** RU-12 is NON-FUNCTIONAL,
   THEN ensure ALL of the following prior to WGDT release:

   - Two independent samples are analyzed
   - Two qualified individuals independently verify the release rate calculations
   - Two qualified individuals independently verify the valve lineups

Technical Reference:	ODCM (Off-site Dose Calculation Manual)

2.0 GASEOUS EFFLUENT MONITOR SETPOINTS

2.1 Requirements: Gaseous Monitors

The radioactive gaseous effluent monitoring instrumentation channels shown in Table 2-1 shall be FUNCTIONAL with their alarm/trip setpoints set to ensure that the dose requirements in Section 3.0 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in Section 2.1.2.

Applicability: As shown in Table 2-1.

Action:

a. With the low range radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Requirement, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel nonfunctional, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels FUNCTIONAL, take the ACTION shown in Table 2-1. Restore the nonfunctional instrumentation to FUNCTIONAL status within 30 days or, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why this nonfunctionality was not corrected within the time specified.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS FUNCTIONAL	APPLICABILITY	ACTION
1. GASEOUS RADWASTE SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release #RU-12	1	#	35
b. Flow Rate Monitor	1	#	36

ACTION 35 - With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tanks contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup;

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Service Water: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits	**Tier**			**2**
	Group			**1**
	K/A	**076 G 2.2.25**		
	IR			**4.2**

Question 90

Per the Technical Specification Bases for LCO 3.7.8, Essential Spray Pond System (ESPS)…

(1) If Train ‘A’ ESPS is isolated from the ‘A’ EDG Coolers, Train ‘A’ ESPS…

(2) If Train ‘A’ ESPS is isolated from the ‘A’ EW Heat Exchanger, Train ‘A’ ESPS …

A. (1) is INOPERABLE
(2) is INOPERABLE

B. (1) is INOPERABLE
(2) remains OPERABLE

C. (1) remains OPERABLE
(2) is INOPERABLE

D. (1) remains OPERABLE
(2) remains OPERABLE

Proposed Answer:	C

Explanations:	
A.	First part is plausible since the ‘A’ EDG is the class backup power source for the ‘A’ ESPS, and the isolation from the EDG coolers would make the EDG inoperable, however it does not render the ESPS inoperable. Second part is correct.
B.	First part is plausible since the ‘A’ EDG is the class backup power source for the ‘A’ ESPS, and the isolation from the EDG cooler would make the EDG inoperable, however it does not render the ESPS inoperable. Second part is plausible since the ESPS is the supporting system in the ESPS/EW relationship, however since the cooling done by the EW system is part of the design basis of the ESPS, the ESPS cannot perform it’s safety function while isolated from the EW system and is therefore inoperable.
C.	Correct.
D.	First part is correct. Second part is plausible since the ESPS is the supporting system in the ESPS/EW relationship, however since the cooling done by the EW system is part of the design basis of the ESPS, the ESPS cannot perform it’s safety function while isolated from the EW system and is therefore inoperable.

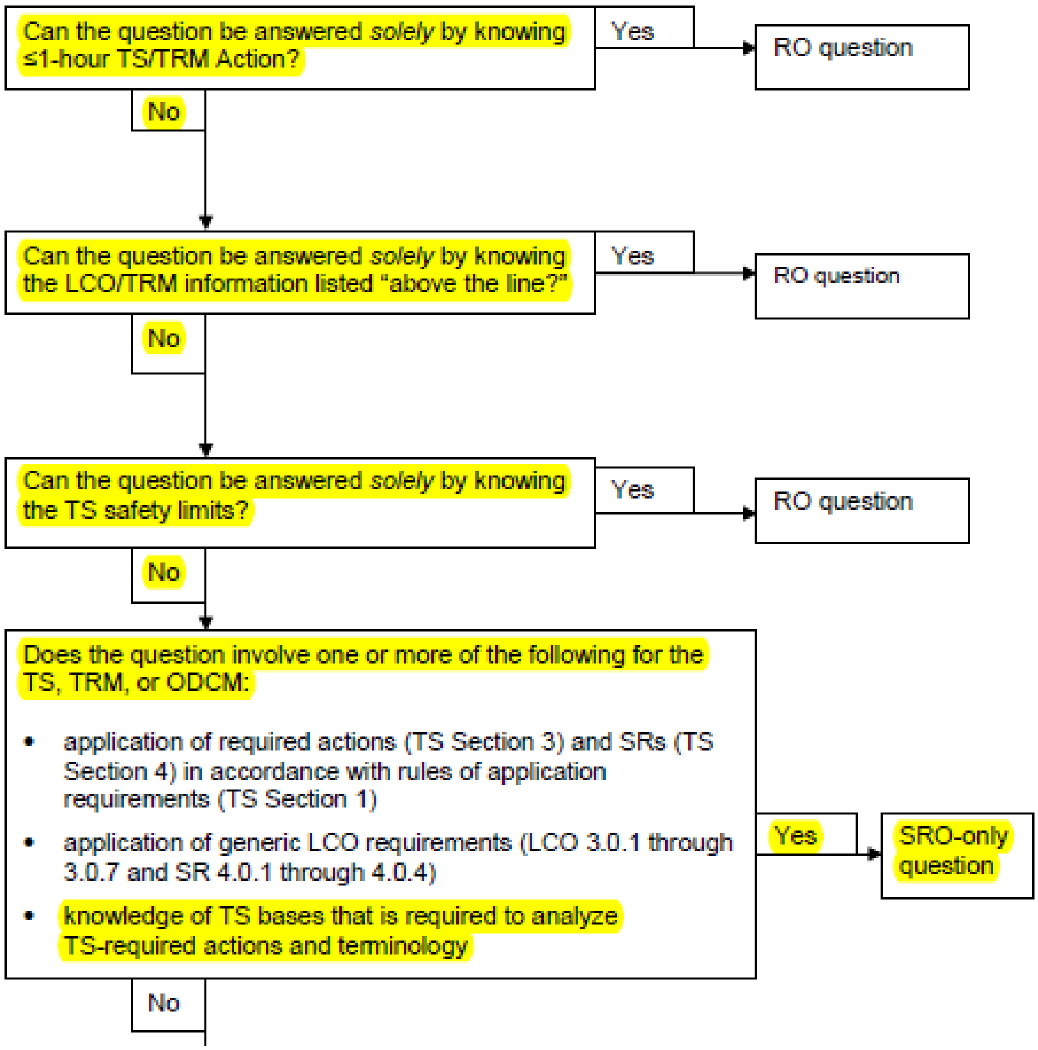
Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	4	
10CFR55.43:	2	
Reference Provided:	N	
Learning Objective:	Given a set of plant conditions, determine whether or not the LCOs and TLCOs of 3.7 are satisfied in accordance with Tech Spec 3.7.	

Technical Reference:	NUREG 1021 SRO-Only Guidance

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Technical Reference:	Technical Specification Bases (LCO 3.7.8)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The ESPS satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).
LCO	Two ESPS trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post-accident heat loads, assuming the worst single active failure occurs coincident with the loss of offsite power. An ESPS train is considered OPERABLE when: a. The associated pump is OPERABLE; and b. The associated piping, valves, instrumentation, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE. The isolation of the ESPS from other components or systems renders those components or systems inoperable, but does not necessarily affect the OPERABILITY of the ESPS. Isolation of the ESPS to required Diesel Generator (DG) cooler(s), while rendering the DG inoperable, is acceptable and does not impact the OPERABILITY of the ESPS. Disassembly, removal of insulation, and other configuration changes to the isolated portions of an OPERABLE system must be explicitly evaluated for operability impact prior to executing any configuration changes of the OPERABLE system. Isolation of the ESPS to the essential cooling water heat exchanger is not acceptable and would render both the Essential Cooling Water System and the ESPS inoperable (Ref. 3). The ESPS is inoperable in this situation because it is operating outside of the acceptable limits of the system.

Examination Outline Cross-Reference:	**Level**	**RO**	[redacted]	**SRO**
K/A: Pressurizer Level Control: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits	**Tier**		[redacted]	**2**
	Group		[redacted]	**2**
	K/A	**011 G 2.2.25**		
	IR		[redacted]	**4.2**

Question 91

Per the Technical Specification Bases for LCO 3.4.9, Pressurizer, the low end of the Pressurizer level requirement is set to ensure ___(1)___ and the high end of the level band is to ensure ___(2)___ .

A. (1) the Pressurizer heaters remain covered
(2) the Main and Auxiliary Spray nozzle is not submerged

B. (1) the Pressurizer heaters remain covered
(2) the proportional heaters can heat the water mass enough to maintain 2250 psia

C. (1) Pressurizer pressure does not lower to the SIAS setpoint on an uncomplicated Reactor trip
(2) the Main and Auxiliary Spray nozzle is not submerged

D. (1) Pressurizer pressure does not lower to the SIAS setpoint on an uncomplicated Reactor trip
(2) the proportional heaters can heat the water mass enough to maintain 2250 psia

Proposed Answer:	**A**

Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible since only the proportional heaters are normally used to maintain pressure while at power, and a larger water mass in the pressurizer could require more (or larger) heaters to maintain 2250 psia, however this is not the basis for the high end of the TS level band.
C.	First part is plausible since
D.	

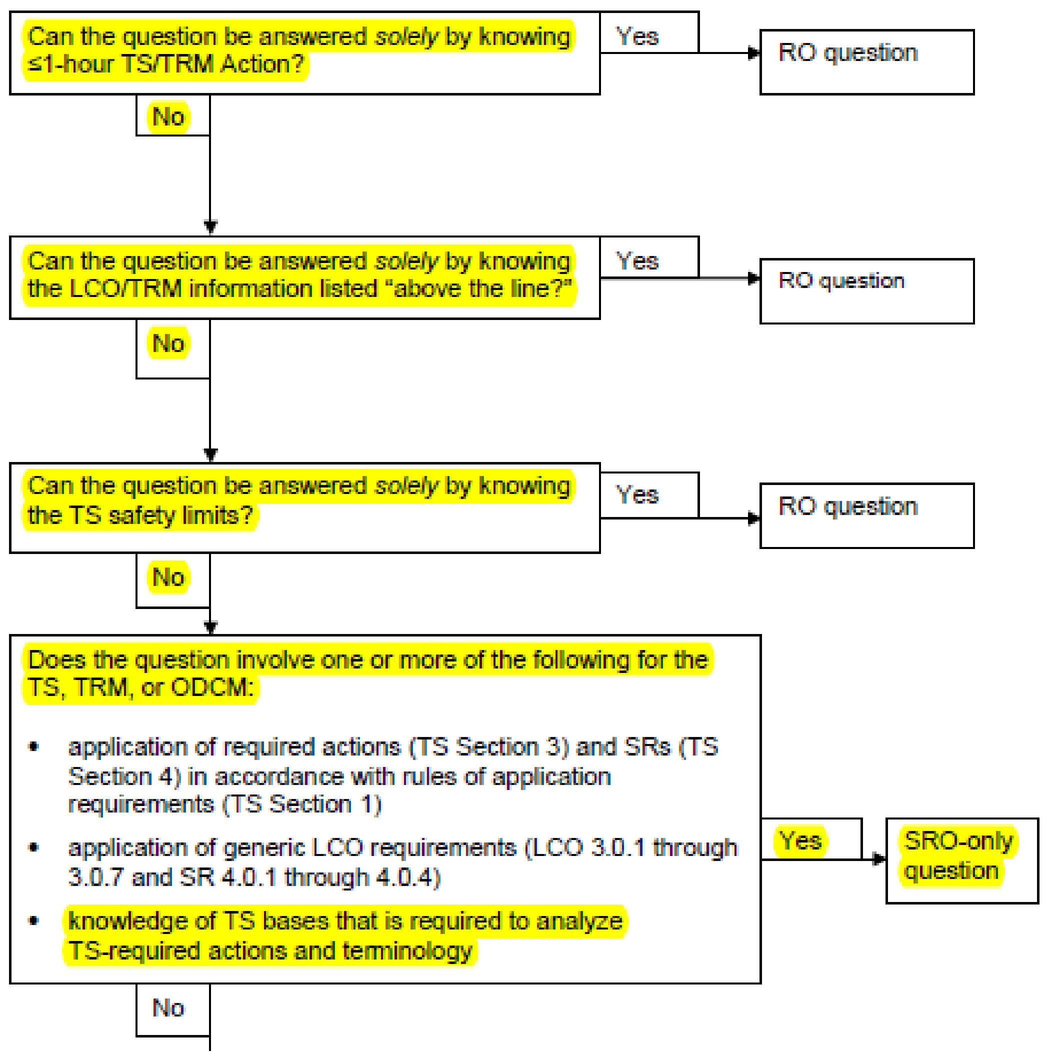
Question Source:		**New**	
	X	**Bank**	
		Modified	
	X	**Previous NRC Exam**	**2016 NRC Q56**

Cognitive Level:	**X**	**Memory or Fundamental Knowledge**
		Comprehension or Analysis

Level of Difficultly:	**2**	
10CFR55.43:	**2**	
Reference Provided:	**N**	
Learning Objective:	**Identify the basis of Technical Specification LCOs and TLCOs for section 3.4 in accordance with Tech Spec 3.4 basis.**	

Technical Reference:	NUREG 1021 SRO-Only Guidance

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Technical Reference:	Technical Specification Bases (LCO 3.4.9)

B 3.4 REACTOR COOLANT SYSTEMS (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level and the required heaters and their backup heater controls. Pressurizer safety valves and pressurizer vents are addressed by LCO 3.4.10 "Pressurizer Safety Valves MODES 1, 2, and 3," LCO 3.4.11 "Pressurizer Safety Valves MODE 4," and LCO 3.4.12 "Pressurizer Vents", respectively.

The maximum steady state water level limit has been established to ensure that a liquid to vapor interface exists to permit RCS pressure control, using the sprays and heaters during normal operation and proper pressure response for anticipated design basis transients. The maximum and minimum steady state water level limit serves two purposes:

a. Pressure control during normal operation maintains subcooled reactor coolant in the loops and thus in the preferred state for heat transport; and

b. By restricting the level to a maximum, expected transient reactor coolant volume increases (pressurizer insurge) will not cause excessive level changes that could result in degraded ability for pressure control.

The maximum steady state water level limit permits pressure control equipment to function as designed. The limit preserves the steam space during normal operation, thus, both sprays and heaters can operate to maintain the design operating pressure. The level limit also prevents filling the pressurizer (water solid) for anticipated design basis transients, thus ensuring that pressure relief devices

BACKGROUND (continued)

(pressurizer safety valves) can control pressure by steam relief rather than water relief. If the level limits were exceeded prior to a transient that creates a large pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the Safety Limit of 2750 psia.

The minimum steady state water level in the pressurizer assures pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized uncovered.

The requirement to have two groups of pressurizer heaters ensures that RCS pressure can be maintained. The pressurizer heaters maintain RCS pressure to keep the reactor coolant subcooled. Inability to control RCS pressure during natural circulation flow could result in loss of single phase flow and decreased capability to remove core decay heat.

Technical Reference:	Technical Specification Bases (LCO 3.4.9)

LCO

The LCO requirement for the pressurizer to be OPERABLE with water level ≥ 27% indicated level (425 cubic feet) and ≤ 56% indicated level (948 cubic feet) ensures that a steam bubble exists. Limiting the maximum operating water level preserves the steam space for pressure control. The LCO has been established to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity ≥ 125 kW. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide subcooling margin to saturation can be obtained in the loops.

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: In-Core Temperature Monitor: Knowledge of system purpose and/or function	**Tier**		[redacted]	**2**
	Group		[redacted]	**2**
	K/A	**017 G 2.1.27**		
	IR		[redacted]	**4.0**

Question 92

Per EP-0801, EAL Hot Chart, a POTENTIAL LOSS of the Fuel Cladding Barrier exists when either…

(1) Representative Core Exit Thermocouple temperature exceeded a MINIMUM of…

OR

(2) Reactor Vessel Level Monitoring System indicates less than a MAXIMUM of…

A. (1) 700°F
   (2) 16% in the upper head

B. (1) 700°F
   (2) 21% in the outlet plenum

C. (1) 1200°F
   (2) 16% in the upper head

D. (1) 1200°F
   (2) 21% in the outlet plenum

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible since this is the level in the vessel below which SI throttle is not allowed, however a potential loss of the fuel cladding barrier doesn't occur until level is less than 21% in the plenum.
B.	Correct.
C.	First part is plausible since 1200°F is the temperature above which the fuel cladding barrier is considered LOST, and is the temperature above which the containment barrier is considered a potential loss. Second part is plausible since this is the level in the vessel below which SI throttle is not allowed, however a potential loss of the fuel cladding barrier doesn't occur until level is less than 21% in the plenum.
D.	First part is plausible since 1200°F is the temperature above which the fuel cladding barrier is considered LOST, and is the temperature above which the containment barrier is considered a potential loss. Second part is correct.

Question Source:		New	
	X	Bank	
		Modified	
	x	Previous NRC Exam	2016 NRC Q91

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	1	
Reference Provided:	N	
Learning Objective:	Determine if any EAL within the Fission Product Barrier category has been met	

Technical Reference:	NUREG 1021 SRO-Only Guidance

Having an Emergency Plan is a condition of license, although not specifically called out by the SRO-Only Guidance in NUREG 1021, and implementation of the Emergency Plan is an SRO-Only job function at PVNGS.

A. Conditions and Limitations in the Facility License [10 CFR 55.43(b)(1)]

Examples of SRO exam items for this topic include the following:

- reporting requirements when the maximum licensed thermal power output is exceeded
- administration of fire protection program requirements, such as compensatory actions associated with inoperable sprinkler systems and fire doors
- required actions necessary when a facility does not meet the administrative controls listed in Technical Specifications (TS), Section 5 or 6, depending on the facility (e.g., shift staffing requirements)
- National Pollutant Discharge Elimination System requirements, if applicable
- processes for TS and final safety analysis report changes

The SRO-only test item is required to be tied to one of the 10 CFR 55.43(b) items. However, if a licensee desires to evaluate a K/A that is not tied to one of the 10 CFR 55.43(b) items, the licensee can classify the K/A as "unique to the SRO position" provided that there is documented evidence that ties the K/A to the licensee's SRO job position duties in accordance with the systematic approach to training.

Justification. A question that is not tied to one of the 10 CFR 55.43(b) items can still be classified as "SRO-only" provided that the licensee has documented evidence to prove that the K/A is "unique to the SRO position" at the site. An example of documented evidence includes the following:

- The question is linked to a learning objective that is specifically labeled in the lesson plan as being SRO-only (e.g., some facility licensee lesson plans have columns in the margin that differentiate auxiliary operator, RO, and SRO learning objectives). [Section D.2.d of this examination standard]

AND/OR

- A question is linked to a task that is labeled as an SRO-only task, and the task is NOT listed in the RO task list.

From PVNGS SRO-Only Master Task List:

MASTER TASK LIST

Task list for OPTRNG at 2020/01/29: (189524) Senior Reactor Operator
All Tasks

Task#	Task	Selected for Training	Recurring	How Often	Training Setting
L392177	Transfer command and control of the Emergency Coordinator functions	Yes	No		Classroom
L392178	Perform the duties of the Emergency Coordinator	Yes	Yes	Yearly	Initial: Classroom & Simulator Continuing: Simulator

Technical Reference:	E-plan EAL Hot Chart

Fuel Clad (FC) Barrier	
Loss	**Potential Loss**
	1. RVLMS < 21% plenum (Detector #8)
1. Rep CETs > 1200ºF	1. Rep CETs > 700ºF 2. RCS heat removal cannot be established **AND** RCS subcooling < 24ºF
1. Containment radiation RU-148 > 2.1E+05 mR/hr **OR** RU-149 > 2.4E+05 mR/hr 2. Dose equivalent I-131 coolant activity > 300 µCi/gm	

Technical Reference:	40EP-9EO10-002 Appendix 2: Figures

16% RVUH level is used for SI throttle criteria and various other checks in the EOPs

SI THROTTLE CRITERIA

CAUTION

Throttling HPSI injection valves will cause erosion damage to downstream piping.

HPSI THROTTLE CRITERIA

- At least one HPSI Pump is operating
- RCS is greater than or equal to 24°F [44°F] subcooled
- Pressurizer level is greater than 10% [15%] and **NOT** lowering
- At least one Steam Generator is available for RCS heat removal with level being maintained within or being restored to 45 - 60% NR [45 - 60% NR]
- RVLMS indicates RVUH level is greater than or equal to 16%
- **IF** the Functional Recovery procedure is in use,
  THEN ensure HPSI Pump(s) are **NOT** being used to meet an RC success path

LPSI THROTTLE CRITERIA

- Pressurizer pressure is greater than 220 psia [220 psia] and is being controlled

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Fire Protection: Ability to (a) predict the impacts of the following malfunctions or operations on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure to actuate the FPS when required, resulting in fire damage	Tier			2
	Group			2
	K/A		086 A2.04	
	IR			3.9

Question 93

Given the following conditions:

- Unit 1 was operating at 100% power when a fire occurred in the Satellite Technical Support Center
- The Fire Suppression System failed to actuate and the CRS has entered 40AO-9ZZ19, Control Room Fire
- The Reactor has been tripped and the crew has evacuated to the Remote Shutdown Panel
- The Emergency Coordinator has declared an ALERT due to the Control Room Evacuation

Per 40AO-9ZZ19, Control Room Fire, and 40DP-9ZZ04, Time Critical Action Program, the CRS must ensure that the ___(1)___ within a MAXIMUM of 5 minutes of the Reactor Trip.

The Emergency Coordinator must ensure State and Local Agencies are notified of the event within a MAXIMUM of ___(2)___ from the time of the ALERT declaration.

A. (1) ADV disconnect switches are placed in LOCAL
   (2) 15 minutes

B. (1) ADV disconnect switches are placed in LOCAL
   (2) 1 hour

C. (1) Letdown to Regen HX Isolation Valve, CHB-UV-515, is closed
   (2) 15 minutes

D. (1) Letdown to Regen HX Isolation Valve, CHB-UV-515, is closed
   (2) 1 hour

Proposed Answer:	A

Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible since the NRC must be notified within a maximum of 1 hour, however state and local agencies must be notified within 15 minutes.
C.	First part is plausible since UV-515 is required to be closed in a finite amount of time per the listed procedures, however the time limit for this closure is 20 minutes. Second part is correct.
D.	First part is plausible since UV-515 is required to be closed in a finite amount of time per the listed procedures, however the time limit for this closure is 20 minutes. Second part is plausible since the NRC must be notified within a maximum of 1 hour, however state and local agencies must be notified within 15 minutes.

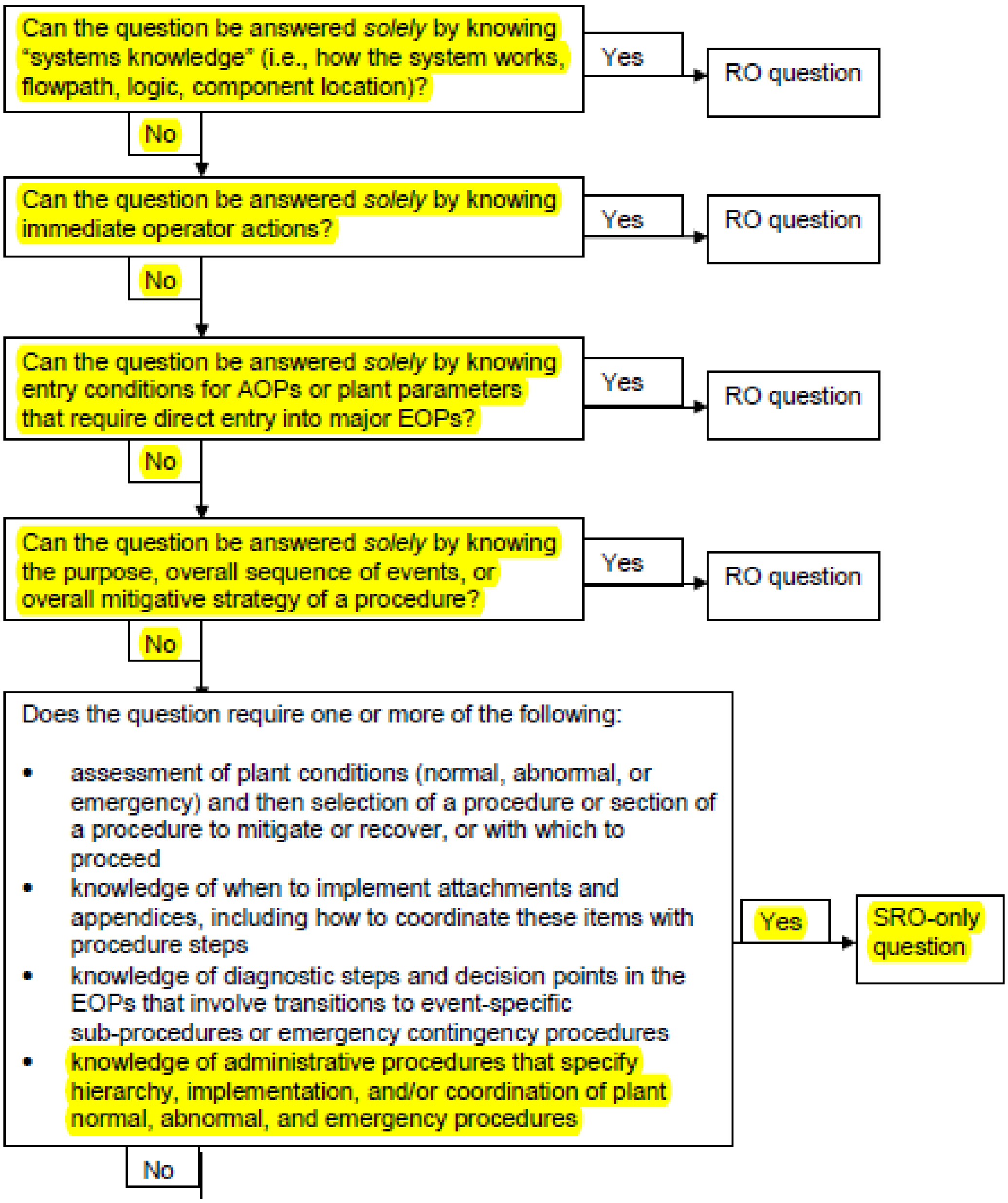
Question Source:		**New**	
	X	**Bank**	
		Modified	
	x	**Previous NRC Exam**	**2018 NRC Q84**

Cognitive Level:	x	**Memory or Fundamental Knowledge**
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	5	
Reference Provided:	N	
Learning Objective:	**For a Control Room Fire, state the Time Critical Actions per 40AO-9ZZ19.**	

Technical Reference:	NUREG 1021 SRO-Only Guidance

**Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)**



Technical Reference:	40DP-9ZZ04 Time Critical Action (TCA) Program

Appendix D - Time Critical Actions Catalog

TCA	Action	Time Limit	Time Zero	Validation Method	Procedure	Org	Source Document (other info)
TCA-1	Initiate MSIS	90 seconds	Manual Reactor Trip	Simulator	40AO-9ZZ19	Ops	13-MC-FP-0318 (40A0-9ZZ19, Section 3.0)
TCA-2	ADV Disconnects to LOCAL	5 minutes	Manual Reactor Trip	Walkdown	40AO-9ZZ19	Ops	13-MC-FP-0318 (40A0-9ZZ19, Section 3.0)
TCA-3	Close CHB-UV-515	20 minutes	Manual Reactor Trip	Walkdown	40AO-9ZZ19	Ops	13-MC-FP-0318 (40A0-9ZZ19, Section 3.0)

Technical Reference:	EP-0902 Notifications

3.2 Limitations

3.2.1 Place keeping within this procedure is conducted using individual ERO Checklists from EP-0900, Emergency Response Organization (ERO) Position Checklists.

3.2.2 Notification to offsite agencies is required within 15 minutes of:

- Initial Classification of the Emergency
- Change in the Classification
- Change in Protective Action Recommendations (PARs)
- Change in Radiological Release Status
- Event Termination

3.2.3 **IF** the primary and alternate communications are not available, **THEN** the following are acceptable communications methods with outside agencies:

- Commercial phone
- Cellular phone
- Satellite phone (Flex/BDBEE Equipment)

3.2.4 The ENS Communicator shall notify the NRC immediately following notifications to the state and county agencies and no later than one hour after the emergency has been declared.

NOTE

- Notification duties and responsibilities transition from the Control Room to the EOF with transfer of Command and Control.
- If the EOF is not activated or is unable to accept Command and Control, the responsibility for required Notification of Offsite Agencies remain in the Satellite Technical Support Center (STSC).
- All of the required offsite state and local agencies must be notified within 15 minutes of determining any of the following criteria:

  • Initial Classification of the Emergency
  • Change in the Classification
  • Change in Protective Action Recommendations (PARs)
  • Change in Radiological Release Status
  • Event Termination

- Responding agencies listed in Section 8, on the EP-541 NAN Form that are BOLDED can be contacted 24hrs/day, UNBOLDED agencies may not answer during non-business hours.

6.4 EP-0541, NAN Emergency Message Form Instructions
Palo Verde forms EP-0541

Examination Outline Cross-Reference:	**Level**	**RO**		**SRO**
K/A: Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc.	**Tier**			**3**
	Group			
	K/A	**G 2.1.4**		
	IR			**3.8**

Question 94

Given the following conditions:

- A licensed SRO has stood the following shifts in the past four months:
  - BOP on night shift on 9/8/2020, 9/9/2020, and 9/10/2020
  - OATC on night shift on 9/30/2020, 10/1/2020, and 10/2/2020
  - STA on day shift on 12/1/2020, and 12/2/2020
  - CRS on day shift on 12/21/2020 and 12/22/2020
- Today is 12/23/2020 and the SRO is trying to determine what he needs to do, if anything, to ensure he has stood enough shifts to be qualified on 1/1/2021
- The Ops Scheduler has informed the SRO that he is needed for shift on 1/1/2021

Assuming all non-shift related requirements for the quarter have been met, which of the following correctly describes the current license status of the SRO on 1/1/2021?

A. The SRO has completed the requisite amount of shift time in the current quarter to stand any RO or SRO position on 1/1/2021

B. The SRO has completed the requisite amount of shift time in the current quarter to stand any RO position on 1/1/2021 but cannot stand an SRO position on 1/1/2021

C. The SRO has NOT completed the requisite amount of shift time in the current quarter to stand any position on 1/1/2021, and must complete a minimum of one 12-hour shift ONLY in an SRO position prior to 1/1/2021 to stand shift on 1/1/2021

D. The SRO has NOT completed the requisite amount of shift time in the current quarter to stand any position on 1/1/2021, and must complete a minimum of one 12-hour shift in EITHER an RO or SRO position prior to 1/1/2021 to stand shift on 1/1/2021

Proposed Answer:	D

Explanations:	
A.	Plausible since the SRO has stood watch in a licensed position, including at least one in an SRO position, in the current quarter, however since the OATC watch stood on 9/30 was not entirely in the fourth calendar quarter, the SRO needs one additional full 12-hour shift in either an RO or SRO position.
B.	Plausible if thought that since the SRO has stood 6 full watches in the fourth quarter that he would be qualified to stand watch on 1/1/2021, but since only 2 (or 4 if thought that the STA watches count as SRO watches) have been stood, that the SRO can stand an RO watch but not an SRO watch, however in order to be active for the following quarter, and SRO must stand a minimum of one 12-hour watch in the CRS or SM position and at least 4 additional watches in either an SRO or RO position.
C.	Plausible if thought that the SRO must stand a total of 5 12-hour SRO shifts in the current quarter to be active in the next quarter, and that the CRS and STA watches all count towards the SRO watchstanding requirement for the fourth quarter, however the STA watches do not count and either an RO or SRO watch will make the SRO active on 1/1/2021.
D.	Correct.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	1	
Reference Provided:	N	
Learning Objective:	Describe operations expectations when it comes to Training & Qualification in accordance with ODP-1, Operations Principles and Standards.	

Technical Reference:	NUREG 1021 SRO-Only Guidance

A. Conditions and Limitations in the Facility License [10 CFR 55.43(b)(1)]

Examples of SRO exam items for this topic include the following:

- reporting requirements when the maximum licensed thermal power output is exceeded
- administration of fire protection program requirements, such as compensatory actions associated with inoperable sprinkler systems and fire doors
- required actions necessary when a facility does not meet the administrative controls listed in Technical Specifications (TS), Section 5 or 6, depending on the facility (e.g., shift staffing requirements)
- National Pollutant Discharge Elimination System requirements, if applicable
- processes for TS and final safety analysis report changes

Technical Reference:	40DP-9OP02, Conduct of Operations

4.8.4.5 An active NRC Operators License exists when an operator has a valid license and has "actively performed" the functions of an operator or senior operator on a minimum of five 12 hour shifts per calendar quarter.

- The number of operator watchstanders required by Technical Specifications is mode dependent. For watch standing proficiency, mode dependency can affect the number of operator proficiency watchstanders allowed.
- 'Actively Performed' means that the operator held a position on the shift crew that required the individual to be licensed as defined in the Technical Specifications and that the operator carried out and was responsible for the duties covered by that position.
- For maintenance of an active SRO license, any shift spent in either the SM or CRS position will be credited.
- An SRO must stand at least one complete shift per calendar quarter in an SRO only supervisory position. The remainder of the shifts required in a calendar quarter may be performed in either a credited SRO or RO position.
- For time to be credited, time must be a continuous shift (that is: 4 hours of watch shift responsibilities spent on each of 3 different shifts does not equal one 12 hour shift).
- Overtime may be credited if the overtime work is in a position appropriately credited for watchstanding proficiency. Overtime as an extra helper after the official watch has been turned over to another watchstander does not count toward proficiency time.
- Counting a shift that will not be completed in the current calendar quarter does not satisfy the requirements of 10 CFR 55.53(e) or NUREG-1021. The last night shift of a quarter actually bridges 2 calendar quarters and therefore, cannot be used for a credited proficiency watch.

Examination Outline Cross-Reference:	**Level**	**RO**		**SRO**
K/A: Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels	**Tier**			**3**
	Group			
	K/A	**G 2.2.2**		
	IR			**4.6**

Question 95

Given the following conditions:

- Unit 2 is commencing a unit startup following a refueling outage
- Preparations are being made to enter MODE 3

Per 40OP-9ZZ11, Mode Change Checklist, entry into MODE 3 should be made within a MAXIMUM of ___(1)___ hours of STARTING the Mode 4 to Mode 3 checklist.

If a piece of TS required equipment was repaired during the refueling outage, is required to be OPERABLE in MODE 3, but retest conditions cannot be established until MODE 3, the MODE change can be performed as allowed by ___(2)___ .

A. (1) 12
(2) SR 3.0.1

B. (1) 12
(2) SR 3.0.3

C. (1) 24
(2) SR 3.0.1

D. (1) 24
(2) SR 3.0.3

Proposed Answer:	C

Explanations:	
A.	First part is plausible since several STs are performed once per shift, so it would be make sense that in order to change modes all STs must be current, however the requirement is to change modes within a maximum of 24 hours of starting the mode change checklist. Second part is correct.
B.	First part is plausible since several STs are performed once per shift, so it would be make sense that in order to change modes all STs must be current, however the requirement is to change modes within a maximum of 24 hours of starting the mode change checklist. Second part is plausible since SR 3.0.3 does allow the deference of STs to a later point in time, however the allowance in SR 3.0.3 only pertains to those STs which are missed, and not due to required test conditions not being met.
C.	Correct.
D.	First part is correct. Second part is plausible since SR 3.0.3 does allow the deference of STs to a later point in time, however the allowance in SR 3.0.3 only pertains to those STs which are missed, and not due to required test conditions not being met.

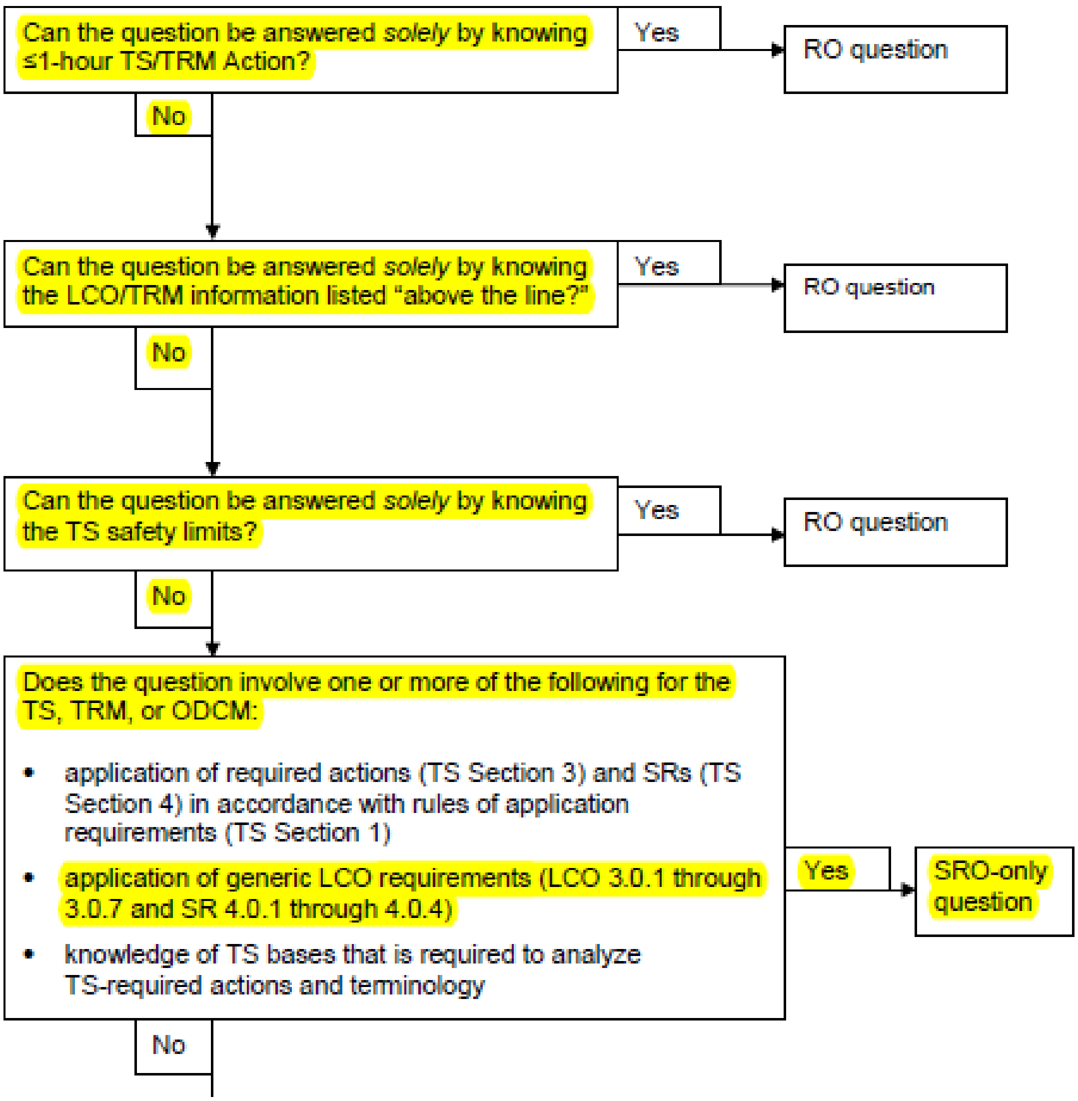
Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	2	
Reference Provided:	N	
Learning Objective:	Concerning Technical Specifications, describe the requirements of SR 3.0.1 in accordance with the Tech Specs.	

Technical Reference:	NUREG 1021 SRO-Only Guidance

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Technical Reference:	Technical Specifications Bases

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified Frequency. Additionally, the definitions related to instrument testing (e.g., CHANNEL CALIBRATION) specify that these tests are preformed by means of any series of sequential, overlapping, or total steps.

SR 3.0.1 (continued)

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

Some examples of this process are:

a. Auxiliary Feedwater (AFW) pump turbine maintenance during refueling that requires testing at steam pressures > 800 psi. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing.

Technical Reference:	Technical Specifications

SR 3.0.3 addresses missed surveillance requirements

SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. The delay period is only applicable when there is a reasonable expectation the surveillance will be met when performed. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
SR 3.0.3 (continued)	When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

Technical Reference:	40OP-9ZZ11 Mode Change Checklist

3.0 PRECAUTIONS AND LIMITATIONS

3.1 Precautions

3.1.1 None

3.2 Limitations

3.2.1 The normal expectation that a Mode/Condition Change checklist be completed and a Mode/Condition Change accomplished within 24 hours of starting the checklist. This ensures BOTH of the following:

- Proper coverage of requirements for the Mode/Condition Change
- Proper awareness of the Mode/Condition Change by all applicable departments

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits	**Tier**			**3**
	Group			
	K/A		**G 2.2.25**	
	IR			**4.2**

Question 96

In which of the following situations, INDIVIDUALLY, can LCO 3.0.6 be invoked to avoid having to comply with the required actions of the supported systems?

1. Train ‘A’ Essential Chiller being declared inoperable due to the Train ‘A’ Spray Pond Pump being out of service for preventive maintenance
2. Train ‘B’ AFW Pump being declared inoperable due to the AFB Pump Room Essential ACU being out of service for preventive maintenance
3. Both Trains of Essential Spray Pond being declared inoperable due to the Ultimate Heat Sink being declared inoperable due to high temperature

A. 2 ONLY

B. 3 ONLY

C. 1 and 2 ONLY

D. 1 and 3 ONLY

Proposed Answer:	D

Explanations:	
A.	Plausible since the room cooler doesn't directly interface with the AF pump as the inop components do in examples 1 and 3, however the inoperability of the room cooler renders the AF Pump inoperable and 3.0.6 is not able to be invoked to avoid the required actions of LCO 3.7.5/
B.	Plausible that since example 3 can utilize the conditions of LCO 3.0.6, however example 1 may also utilize LCO 3.0.6.
C.	Plausible since condition 1 may use LCO 3.0.6, however condition 2 cannot since the support piece of equipment which makes the AF Pump inoperable does not have its own TS, therefore not taking the actions of the AF Pump LCO would essentially be entering no required actions.
D.	Correct.

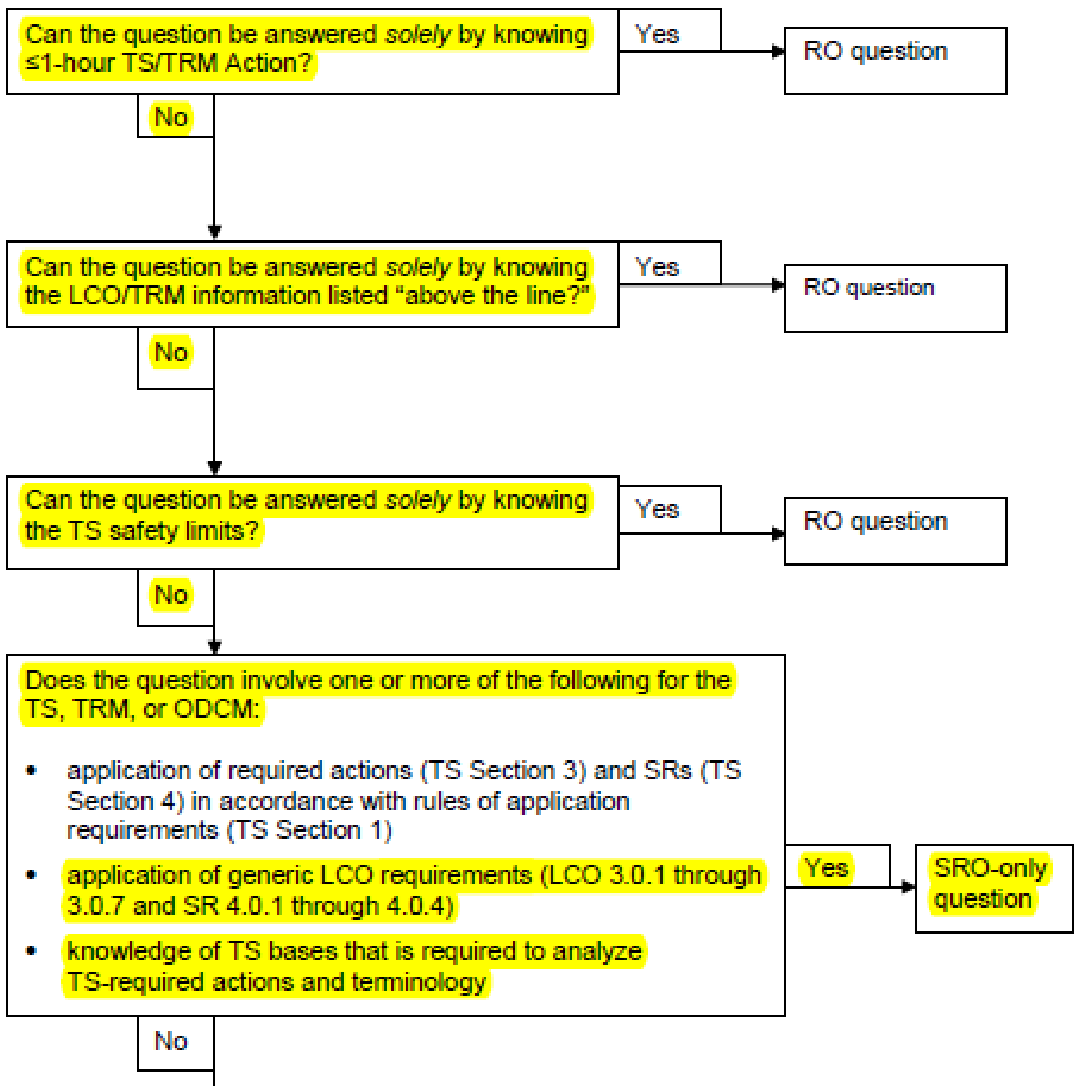
Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	4	
10CFR55.43:	2	
Reference Provided:	N	
Learning Objective:	Concerning Technical Specifications, describe the requirements of LCO 3.0.6 in accordance with the Tech Specs.	

Technical Reference:	NUREG 1021 SRO-Only Guidance

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Technical Reference:	Tech Spec Bases

BASES

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

Technical Reference:	Technical Specifications (TS 5.5.15 SFDP)

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

5.5.15 Safety Function Determination Program (continued)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

Technical Reference:	Technical Specification Bases

BASES

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Technical Reference:	Technical Specification Bases

LCO 3.0.6 (continued)

Specification 5.5.15, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. A loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or (EXAMPLE B3.0.6-1)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or (EXAMPLE B3.0.6-2)

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable. (EXAMPLE B3.0.6-3)

If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operation is being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY).

LCO 3.0.6 (continued)

When a loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately addresses the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

Technical Reference:	40DP-9OP37 Safety Function Determination Procedure

One train ESPS (Spray Pond Pump) being inoperable results in the associated train of EC being inoperable. LCO 3.0.6 can be used.

4.1.6 When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

LCO	SUPPORT FEATURE	SUPPORTED LCO
3.7.7	EW	3.4.6 RCS Loops - Mode 4 3.7.10 EC (40ST-9EC03)*
3.7.8	ESPS	3.4.6 RCS Loops - Mode 4 3.7.7 EW* 3.8.1 AC Sources - Operating (40ST-9ZZ37)
3.7.9	UHS (Note 1)	3.7.8 ESPS*
Note 1	Loss of the single support systems may render multiple trains of supported systems inoperable, resulting in a loss of safety function. In those situations, the supported equipment should be declared inoperable, but the Conditions and Required Actions only for the support systems (RWT, CST, UHS) are required to be entered.	

Example 2 is for a room cooler being non-functional (similar to an AFW pump room cooler). Since the room cooler is not in Tech Specs, LCO 3.0.6 cannot be used. LCO 3.0.6 can only be used if an LCO support system is inoperable.

Example # 2 - Support Feature that is not specified in Technical Specification

For this example HAA-Z01, HPSI Pump A Room Essential ACU, is considered inoperable due to a bad fan bearing. HAA-Z01 is not specified by an LCO in the Technical Specification, thus the first step in determining if LCO 3.0.6 could be implemented would not be met. Since HPSI Pump A is specified in Technical Specification, the LCO Required Actions would be performed for HPSI Pump A. (3.5.3 ECCS - Operating).

X = Inoperable Support Feature / = Inoperable Supported Feature (support features are on the left)

Example 3 shows how the CST being inoperable makes all three trains of AFW inoperable resulting in a loss of safety function. However, the CST is a single support system and LCO 3.0.6 can be used. UHS is also a single support system that results in a loss of safety function.

Example # 3

Appendix D, SUPPORT & SUPPORTED LCO MATRIX

LCO	SUPPORT FEATURE	SUPPORTED LCO
3.7.6	CST	3.7.5 AFW

Appendix D indicates that the AFW LCO 3.7.5 is considered a supported LCO of the CST LCO 3.7.6. Since the CST serves all three AFW Pumps, all of the AFW Pumps are considered to be inoperable.

By determining the impact of the inoperable support feature early in the process, a potential loss of safety function associated with the AFW Pumps has been identified. The Required Actions in LCO 3.7.6 for the CST, if completed within the allowed Completion Time for LCO 3.7.6, should remedy the loss of safety function associated with the AFW Pumps. Also, because this loss of safety function would be due to a single support system, then only the Conditions and Required Actions that need to be entered are those of the CST.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	**Tier**			**3**
	Group			
	K/A	**G 2.3.15**		
	IR			**3.1**

Question 97

While evaluating the PAR Flowchart, which of the following Radiation Monitors are used to determine if a Rapidly Progressing Severe Accident is in progress?

1. Containment Radiation Monitors, RU-148 and 149
2. Main Steam Line Radiation Monitors, RU-139 and 140
3. Plant Vent Radiation Monitors, RU-143 and 144
4. Fuel Building Exhaust Radiation Monitors, RU-145 and 146

A. 1 ONLY

B. 1 and 2 ONLY

C. 2, 3, and 4 ONLY

D. 3 and 4 ONLY

Proposed Answer:	A

Explanations:	
A.	Correct.
B.	Plausible that since the PAR is based largely on the amount of release to the environment that both releases inside and outside of containment would be evaluated (i.e. LOCA and SGTR), however only the Containment High Range monitors are used when determining if the event is rapidly progressing.
C.	Plausible that the MSL, PV and FB monitors would all be evaluated since these encompass three areas in which radiation can escape via monitored pathways, however MSL monitors are not used and the other two are used only for the release flowchart as they are the effluent RMs.
D.	Plausible since these RMs are the ones used when determined the status of a release as they are the credited effluent RMs, however they are not used when making protective action recommendations.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	4	
Reference Provided:	N	
Learning Objective:	Describe terms and requirements associated with Protective Actions	

Technical Reference:	NUREG 1021 SRO-Only Guidance

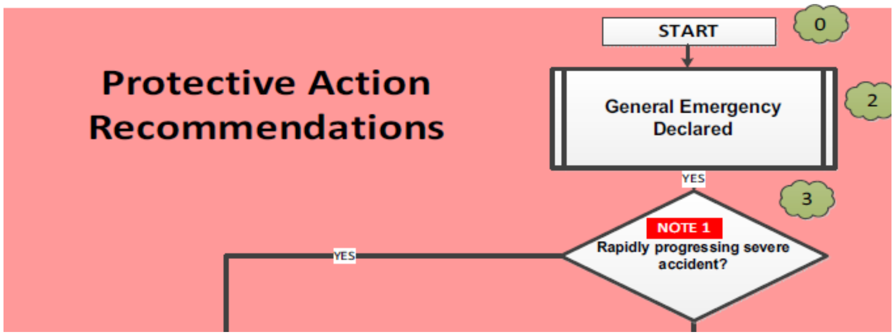
D. Radiation Hazards That May Arise during Normal and Abnormal Situations, including Maintenance Activities and Various Contamination Conditions [10 CFR 55.43(b)(4)]

Some examples of SRO exam items for this topic include the following:

- process for gaseous/liquid release approvals (i.e., release permits)
- analysis and interpretation of radiation and activity readings as they pertain to the selection of administrative, normal, abnormal, and emergency procedures
- analysis and interpretation of coolant activity, including comparison to emergency plan criteria and/or regulatory limits

Technical Reference:	EP-0905 Protective Actions

RU-148/149 used for determining a Rapidly Progressing Severe Accident.



NOTE 1 **Rapidly Progressing Severe Accident**

1. This is the <u>initial</u> **PAR** after a General Emergency has been declared
   AND
2. There is LOSS of the containment barrier per the Emergency Action Levels
   AND
3. EITHER of the following:
   a. Containment High Range Rad monitor reading greater than or equal to:
      i. 6.8E+06 mr/hr (RU-148), **OR**
      ii. 7.8E+06 mr/hr (RU-149)
      OR
   b. A significant radiological release (greater than1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the site boundary) is occurring or is projected to occur in less than an hour

NOTE 1 **Rapidly Progressing Severe Accident**

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   AND
2. There is LOSS of the containment barrier per the Emergency Action Levels
   AND
3. EITHER of the following:
   a. Containment High Range Rad monitor reading greater than or equal to:
      i. 6.8E+06 mr/hr (RU-148), **OR**
      ii. 7.8E+06 mr/hr (RU-149)
      OR
   b. A significant radiological release (greater than1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the site boundary) is occurring or is projected to occur in less than an hour

A rapidly progressing severe accident is a General Emergency with rapid loss of containment integrity and loss of ability to cool the core. Note 1 contains a list of specific conditions that are necessary to meet the “rapidly progressing severe accident” threshold. The loss of containment is defined in the Emergency Action Levels. Loss of the ability to cool the core is indicated by a large source term in containment as denoted by high radiation monitor values or a dose assessment indicating a release greater than 1000 mRem TEDE (Total Effective Dose Equivalent) or 5000 mRem Thyroid CDE (Committed Dose Equivalent) at or beyond the site boundary.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	**Tier**			**3**
	Group			
	K/A	**G 2.3.13**		
	IR			**3.8**

Question 98

Per EP-0905, Protective Actions, the EPA guidance for life-saving or protection of large populations allows for a MAXIMUM TEDE exposure of ___(1)___ and authorization for this exposure is required to be given by the ___(2)___ .

A. (1) 10 REM
(2) Emergency Coordinator

B. (1) 10 REM
(2) Radiation Protection Coordinator

C. (1) 25 REM
(2) Emergency Coordinator

D. (1) 25 REM
(2) Radiation Protection Coordinator

Proposed Answer:	**C**

Explanations:	
A.	First part is plausible since this is the limit for protection of property, however 25 REM is the limit for lift saving efforts. Second part is correct.
B.	First part is plausible since this is the limit for protection of property, however 25 REM is the limit for lift saving efforts. Second part is plausible since the Radiation Protection Coordinator is the person who authorizes exposures up to 5 REM, however the EC authorizes exposures from 5 – 25 REM.
C.	Correct.
D.	First part is correct. Second part is plausible since the Radiation Protection Coordinator is the person who authorizes exposures up to 5 REM, however the EC authorizes exposures from 5 – 25 REM.

Question Source:		**New**	
	x	**Bank**	
		Modified	
	x	**Previous NRC Exam**	**2018 NRC Q98**

Cognitive Level:	**X**	**Memory or Fundamental Knowledge**
		Comprehension or Analysis

Level of Difficultly:	**2**	
10CFR55.43:	**4**	
Reference Provided:	**N**	
Learning Objective:	**Identify the EC responsibilities associated with authorizing emergency exposure**	

Technical Reference:	NUREG 1021 SRO-Only Guidance

D. Radiation Hazards That May Arise during Normal and Abnormal Situations, including Maintenance Activities and Various Contamination Conditions [10 CFR 55.43(b)(4)]

Some examples of SRO exam items for this topic include the following:

- process for gaseous/liquid release approvals (i.e., release permits)
- analysis and interpretation of radiation and activity readings as they pertain to the selection of administrative, normal, abnormal, and emergency procedures
- analysis and interpretation of coolant activity, including comparison to emergency plan criteria and/or regulatory limits

Technical Reference:	EP-0905 Protective Actions

6.8 Emergency Exposure Authorization

6.8.1 Determine the exposure limit for the job assignment based on expected radiological conditions using Table 1.

Table 1: Exposure Limits

Dose Limits	TEDE	TEDE & Thyroid CDE	LDE	SDE	Authorization Required By:
10 CFR 20.1201 Limits (EPA guidance for all workers in emergencies)	≤ 5 REM per Year	≤ 50 REM per Year	≤ 15 REM per Year	≤ 50 REM per Year	RPM or RPC
EPA Guidance for Protecting Property*	5-10 REM per Event	50-100 REM per Event	15-30 REM per Event	50-100 REM per Event	EC-STSC or EC-TSC only (when lower limits are not practicable)
EPA Guidance for Life-Saving or Protection of Large Populations**	≤ 25 REM per Event	≤ 250 REM per Event	≤ 75 REM per Event	≤ 250 REM per Event	EC-STSC or EC-TSC only (when lower limits are not practicable)
EPA Guidance for Life-Saving or Protection of Large Populations** (on a Voluntary Basis Only)	> 25 REM per Event	> 250 REM per Event	>75 REM per Event	>250 REM per Event	EC-STSC or EC-TSC only (and a risk discussion must be conducted)
"Protecting Valuable Property" includes equipment-saving measures or repair activities that, in the opinion of the EC, constitutes plant protective measures.					
*"Protection of Large Populations" refers to situations where the collective dose avoided by the emergency operation is significantly larger than that incurred by the workers involved.					

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Knowledge of the specific bases for EOPs	Tier			3
	Group			
	K/A	G 2.4.18		
	IR			4.0

Question 99

Per 40DP-9AP16, EOP Users Guide…

(1) Bulleted lists of options to mitigate an event in progress (i.e. potential methods to restore feedwater) are…

(2) Implementation of an AOP concurrently with the EOP in use may be done…

A. (1) all equally acceptable to pursue
   (2) at the discretion of the CRS

B. (1) all equally acceptable to pursue
   (2) ONLY with the concurrence of the SM

C. (1) listed in the order in which they are REQUIRED to be pursued
   (2) at the discretion of the CRS

D. (1) listed in the order in which they are REQUIRED to be pursued
   (2) ONLY with the concurrence of the SM

Proposed Answer:	A

Explanations: **Given that this is a Tier 3 generic KA, we felt that the best way to match the spirit of the KA while maintaining as a Tier 3 generic question was to ask about the rules of use for the EOPs.**	
A.	Correct.
B.	First part is correct. Second part is plausible since the SM is required to concur if steps are going to be performed other than written or if performing actions outside of the EOPs is deemed necessary, however use of alternate procedures during EOP usage can be done at the discretion of the CRS.
C.	First part is plausible since the methods presented in lists are generally listed in order of preference, however the use of any of the listed methods are all equally acceptable and may be pursued at the discretion of the CRS. Second part is correct.
D.	First part is plausible since the methods presented in lists are generally listed in order of preference, however the use of any of the listed methods are all equally acceptable and may be pursued at the discretion of the CRS. Second part is plausible since the SM is required to concur if steps are going to be performed other than written or if performing actions outside of the EOPs is deemed necessary, however use of alternate procedures during EOP usage can be done at the discretion of the CRS.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	**X**	**Memory or Fundamental Knowledge**
		Comprehension or Analysis

Level of Difficultly:	**3**	
10CFR55.43:	**1**	
Reference Provided:	**N**	
Learning Objective:	**Given that an ORP is being implemented, describe the order in which procedural steps are performed during the performance of the EOP in accordance with 40DP-9AP16.**	

Technical Reference:	NUREG 1021 SRO-Only Guidance

Directing actions from the EOPs (and therefore knowledge of rules of use for EOPs) is an SRO-Only Job Function at PVNGS

A. Conditions and Limitations in the Facility License [10 CFR 55.43(b)(1)]

Examples of SRO exam items for this topic include the following:

- reporting requirements when the maximum licensed thermal power output is exceeded
- administration of fire protection program requirements, such as compensatory actions associated with inoperable sprinkler systems and fire doors
- required actions necessary when a facility does not meet the administrative controls listed in Technical Specifications (TS), Section 5 or 6, depending on the facility (e.g., shift staffing requirements)
- National Pollutant Discharge Elimination System requirements, if applicable
- processes for TS and final safety analysis report changes

The SRO-only test item is required to be tied to one of the 10 CFR 55.43(b) items. However, if a licensee desires to evaluate a K/A that is not tied to one of the 10 CFR 55.43(b) items, the licensee can classify the K/A as "unique to the SRO position" provided that there is documented evidence that ties the K/A to the licensee's SRO job position duties in accordance with the systematic approach to training.

Justification. A question that is not tied to one of the 10 CFR 55.43(b) items can still be classified as "SRO-only" provided that the licensee has documented evidence to prove that the K/A is "unique to the SRO position" at the site. An example of documented evidence includes the following:

- The question is linked to a learning objective that is specifically labeled in the lesson plan as being SRO-only (e.g., some facility licensee lesson plans have columns in the margin that differentiate auxiliary operator, RO, and SRO learning objectives). [Section D.2.d of this examination standard]

AND/OR

- A question is linked to a task that is labeled as an SRO-only task, and the task is NOT listed in the RO task list.

From the PVNGS SRO-Only Master Task List:

MASTER TASK LIST

Task list for OPTRNG at 2020/01/29: (189524) Senior Reactor Operator
All Tasks

Task#	Task	Selected for Training	Recurring	How Often	Training Setting
	SB: Core Protection Calculator				
1060010402	Direct operations with CPC out of service for low power or reactor shutdown	Yes	No		Classroom
	EM - Emergency Operating Procedures				
124000379337	Respond to an event requiring entry into the Severe Accident Management Guidelines	Yes	Yes	4 Years	Initial: Classroom Continuing: Classroom
L498331	Direct actions from the Emergency Operating Procedures	Yes	Yes	Yearly	Initial: Classroom & Simulator Continuing: Simulator

Technical Reference:	40DP-9AP16 Emergency Operating Procedure Users Guide

4.10 STEP SEQUENCING

4.10.1 Steps are sequenced to provide a systematic path to stabilize the plant. The operators follow the steps as written.

4.10.5 Lists are provided within a step when any one of several alternative actions are equally acceptable to perform.

4.18.7 Performing Steps in the EOPs

A. If the left column can not be completed, the user shall move to the right column for contingency actions. If the contingency action can not be accomplished or is not provided, then the user proceeds to the next step in the left column.

B. If the contingency action achieves the expected response, the user proceeds to the next sequential step in the left column unless referred to another step or procedure.

C. The operator shall perform the procedure action steps as written and in the order written. However, a Trigger Step may be performed anytime that the trigger condition becomes true.

D. Emergencies may not proceed as expected. Sufficient flexibility must be provided to aid the CRS with steps that can not be performed as written. If a step can not be performed as written, and the CRS wants to perform the step in another manner, then he shall obtain the concurrence of the SM.

Technical Reference:	40EP-9EO06 Loss of All Feedwater

This EOP step is an example of a list of actions. The best option is to be performed / selected by the CRS.

INSTRUCTIONS

6. Restore feed to at least one Steam Generator using **ANY** of the following:

AUXILARY FEEDWATER

- Appendix 38, Resetting AFA-P01
- Appendix 39, Local Operation of AFB-P01
- Appendix 40, Local Operation of AFA-P01 Using Main Steam
- Appendix 41, Local Operation of AFN-P01
- Appendix 42, Aligning Aux Feedwater Pumps Suction to RMWT
- Appendix 112, Manual Operation of AFA-P01 During a Security Event

MAIN FEEDWATER

- Appendix 43, Restarting MFPs

CONTINGENCY ACTIONS

6.1 Perform the following to establish a low pressure feedwater source:

a. **IF ALL** of the following:

- Auxiliary or Main Feedwater can **NOT** be restored
- Offsite power is available
- Feeding a Steam Generator with a condensate pump is desired

THEN PERFORM Appendix 44, Feeding with the Condensate Pumps.

b. **IF** feeding a Steam Generator with a fire pump is desired, **THEN** PERFORM Appendix 118, Cross-connect FP to AF.

6.2 **IF** feed to at least one Steam Generator can **NOT** be restored, **THEN** GO TO 40EP-9EO09, Functional Recovery to perform **ANY** of the following:

Step 24 of the LOAF EOP does not contain a bulleted list. These sub steps are required to be performed in the order written.

* 24. **IF** SIAS has actuated, **THEN** perform the following:

a. Close the LPSI Injection Valves.

b. Stop LPSI Pumps.

c. Stop CS Pumps.

d. Energize SIAS Load Shed Panels. REFER TO Appendix 21, List of SIAS Load Shed Panels.

e. PERFORM Appendix 17, Restoration of Containment Cooling.

Technical Reference:	40DP-9AP16 Emergency Operating Procedure Users Guide

4.12 REFERENCES TO OTHER PROCEDURES

4.12.1 The EOPs are designed to minimize the interface with other procedures. The EOPs include standard appendices that operators may need during recovery. Minimizing the interface with other procedures and including standard appendices provide the control room staff with an easily located set of instructions while minimizing the simultaneous use of other procedures. When other procedures are needed, the required procedure will be referenced in the body of the EOP.

4.12.2 Other procedures (ALs, AOs, or OPs) are not normally needed to supplement the EOPs, but may be used when directed by the CRS.

4.25 USE OF ABNORMAL AND OPERATING PROCEDURES

NOTE

Performance of an AOP may be in progress when the reactor trips and EOPs are entered.

4.25.1 Reactivity Control Safety Function shall be addressed immediately after a reactor trip, however, some operations in progress will require that additional steps in an AOP be performed prior to addressing additional safety functions (for example, stopping Reactor Coolant Pumps and isolating seal bleedoff).

4.25.2 Continued performance of non-EOP activities should be limited to activities essential for equipment protection, personnel safety, and placing plant systems in a safe condition.

4.25.3 The CRS may direct performance of an AO or OP with an optimal recovery procedure, the FRP, or the LMFR. Use of an AO or OP may be necessary to prevent compounding of an event, such as degraded electrical conditions or a loss of instrument air event.

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations	**Tier**		[redacted]	**3**
	Group		[redacted]	
	K/A	**G 2.4.23**		
	IR		[redacted]	**4.4**

Question 100

Using the Safety Function Tracking Sheet on the following page:

(1) The first Safety Function performed will be…

(2) After all Challenged and Jeopardized Safety Functions are performed, the next Success Path in use to be verified will be…

A. (1) Pressure Control
(2) Reactivity Control

B. (1) Pressure Control
(2) MVDC

C. (1) Heat Removal
(2) Reactivity Control

D. (1) Heat Removal
(2) MVDC

4.0 SAFETY FUNCTION TRACKING

EOP Entry Time __________

Safety Function	Success Path	Path in use	Challenged	Jeopardized	Completed
RC	RC-1; CEA Insertion				
	RC-2; CVCS Boration	✓			
	RC-3; HPSI Boration				
MVDC	MVDC-1; Batt Chargers/Station Batt	✓			
MVAC	MVAC-1; Offsite Power				
	MVAC-2; DGs	✓			
	MVAC-3; SBOGs				
	MVAC-4; Other Unit DGs				
IC	IC-1; CVCS	✓			
	IC-2; SI				
PC	PC-1; Subcooled Pressure Control		✓		
	PC-2; RCGVS				
	PC-3; Saturated Pressure Control				
HR	HR-1; SG with no SI			✓	
	HR-2; SG with SI				
CI	CI-1; Auto/Man CTMT Isolation	✓			
CTPC	CTPC-1; CTMT Fans	✓			
	CTPC-2; CS				

Proposed Answer:	C

Explanations:	
A.	First part is plausible since Pressure Control is the highest SF that is challenged or jeopardized, however the highest jeopardized SF is addressed first. Second part is correct.
B.	First part is plausible since Pressure Control is the highest SF that is challenged or jeopardized, however the highest jeopardized SF is addressed first. Second part is plausible since RC is generally not re-addressed if initially met, however this is only true if RC is met due to all CEAs being fully inserted, not when RC is met due to a boration in progress.
C.	Correct.
D.	First part is correct. Second part is plausible since RC is generally not re-addressed if initially met, however this is only true if RC is met due to all CEAs being fully inserted, not when RC is met due to a boration in progress.

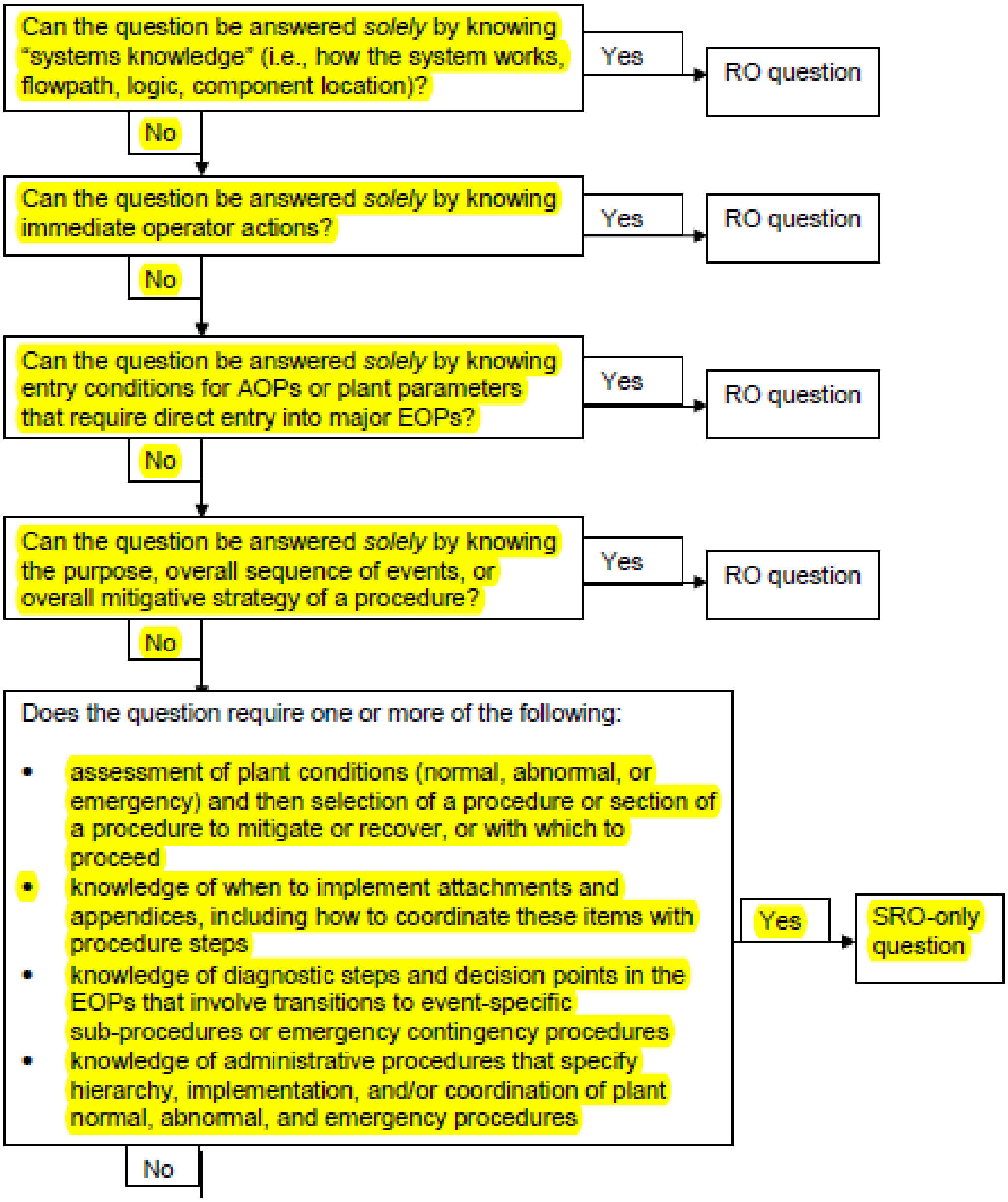
Question Source:		**New**	
		Bank	
	x	**Modified**	
	x	**Previous NRC Exam**	**2020 NRC Q94 (modified)**

Cognitive Level:		**Memory or Fundamental Knowledge**
	x	**Comprehension or Analysis**

Level of Difficultly:	**3**	
10CFR55.43:	**5**	
Reference Provided:	**N**	
Learning Objective:	**Given the FRP is being performed and various combinations of success paths have been identified and jeopardized (if appropriate) , describe the order and the process in which a specific set of success paths will be addressed in accordance with 40EP-9EO9.**	

Technical Reference:	NUREG 1021 SRO-Only Guidance

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Original Question:	2020 NRC Exam Q94

Question 94

Using the Safety Function Tracking Sheet on the following page:

(1) The first Safety Function performed will be…

(2) After all Challenged and Jeopardized Safety Functions are performed, the next Success Path in use to be verified will be…

A. (1) Pressure Control
(2) MVDC

B. (1) Pressure Control
(2) Reactivity Control

C. (1) Heat Removal
(2) MVDC

D. (1) Heat Removal
(2) Reactivity Control

PALO VERDE NUCLEAR GENERATING STATION FUNCTIONAL RECOVERY	40EP-9EO09 Revision 62 Page 8 of 245

4.0 SAFETY FUNCTION TRACKING

EOP Entry Time ____

Safety Function	Success Path	Path in use	Challenged	Jeopardized	Completed
RC	RC-1; CEA Insertion	X			
	RC-2; CVCS Boration				
	RC-3; HPSI Boration				
MVDC	MVDC-1; Batt Chargers/Station Batt	X			
MVAC	MVAC-1; Offsite Power	X			
	MVAC-2; DGs				
	MVAC-3; SBOGs				
	MVAC-4; Other Unit DGs				
IC	IC-1; CVCS	X			
	IC-2; SI				
PC	PC-1; Subcooled Pressure Control	X	X		
	PC-2; RCGVS				
	PC-3; Saturated Pressure Control				
HR	HR-1; SG with no SI	X		X	
	HR-2; SG with SI				
CI	CI-1; Auto/Man CTMT Isolation	X			
CTPC	CTPC-1; CTMT Fans	X			
	CTPC-2; CS				

Technical Reference:	**40EP-9EO09 Functional Recovery**
INSTRUCTIONS ____ 9. Perform **ALL** of the following in the order listed: • Success path instructions for those safety functions that are in jeopardy • Success path instructions for those safety functions that are challenged • Success path instructions for all other non-shaded success paths in use	CONTINGENCY ACTIONS

Technical Reference:	40DP-9AP14 Functional Recovery Technical Guideline

4.2 Procedure Strategy

4.2.1 The basic strategy of the Functional Recovery procedure is to first determine the status of all of the safety functions and then build a procedure using the appropriate success paths that will recover or maintain the acceptance criteria of each safety function.

Once the Functional Recovery procedure has been entered, the operator will use the Safety Function Tracking page and the Resource Assessment Trees (RATs) to determine the acceptance criteria and the equipment needed to satisfy each safety function. The operator must determine whether each safety function is jeopardized (acceptance criteria not met), challenged (acceptance criteria met but action must be taken to ensure that the criteria continue to be met) or satisfied in order to set the priorities for performance of the procedure.

Jeopardized safety functions are addressed first. Challenged safety functions are addressed next, with appropriate actions for satisfied safety functions taken last. All safety functions are addressed in the established hierarchy.

Technical Reference:	40DP-9AP14 Functional Recovery Technical Guideline

4.5.9 Step 9 - Perform Success Paths

A. The intent of this step is to establish a priority for operator actions as the Functional Recovery is implemented.

In the event any safety function SFSC acceptance criteria are not met, the operator's first priority is to perform the operator actions for those success paths that will restore the safety function.

After the actions have been performed for jeopardized safety functions, the appropriate actions for all other success paths in use must be performed. Challenged safety functions are to be addressed next. Challenged safety functions are those which are currently satisfied but may become jeopardized due to plant trends or equipment conditions.

The operator actions associated with the remaining success paths in use contain guidance to enable those safety functions to continue to satisfy their acceptance criteria.

4.6 Safety Function Tracking

4.6.1 The Safety Function Tracking page organizes and condenses information about the success paths in use. It was created to give the CRS a convenient place to keep track of selected success paths in use and their status. The first two columns identify all the success paths as described in section 5.0, Safety Function Status Check. The third column provides the CRS a place to annotate the success paths in use. Also, the third column provides a place to annotate a new success path in use when conditions warrant selection of a new success path. The fourth column provides the CRS with a place to annotate whether the selected success path is challenged. The fifth column provides the CRS with a place to annotate whether the selected success path is in jeopardy or not. The sixth column provides the CRS with a place to annotate that appropriate instructions within the selected success path in use have been completed.

The grayed Completed blocks have a special meaning. Performing the instructions/contingencies for these paths is not required when the associated success path acceptance criteria are satisfied. Meeting the acceptance criteria for these paths implies that no further actions are needed.