U.S. Nuclear Regulatory Commission
Senior Reactor Operator Written Examination

Applicant Information	
Name: **ANSWER KEY**	
Date: 08/11/2023	Facility/Unit: Diablo Canyon Power Plant
Region: I ☐ II ☐ III ☐ IV ☒	Reactor Vendor/Type: Westinghouse
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination, you must achieve a final grade of at least 80 percent overall, with 70 percent or better on the senior reactor operator (SRO)-only items if given in conjunction with the reactor operator (RO) exam; SRO-only exams given alone require a final grade of 80 percent to pass. You have 9 hours to complete the combined examination and 3 hours if you are only taking the SRO-only portion.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results	
RO/SRO-Only/Total Examination Points	______ / ______ / ______ Points
Applicant's Points	______ / ______ / ______ Points
Applicant's Grade	______ / ______ / ______ Percent

Multiple Choice (Fill In Your Choice) NAME: ______**KEY**_____
If you change your answer, write your selection in the blank and initial.

1.	A	B	C	●	____
2.	A	B	●	D	____
3.	A	●	C	D	____
4.	A	●	C	D	____
5.	A	●	C	D	____
6.	●	B	C	D	____
7.	A	B	●	D	____
8.	A	●	C	D	____
9.	●	B	C	D	____
10.	A	B	C	●	____
11.	●	B	C	D	____
12.	A	B	C	●	____
13.	A	B	●	D	____
14.	●	B	C	D	____
15.	A	B	C	●	____
16.	A	B	C	●	____
17.	A	B	●	D	____
18.	A	B	●	D	____
19.	A	●	C	D	____
20.	A	●	C	D	____
21.	●	B	C	D	____
22.	A	B	C	●	____
23.	A	B	●	D	____
24.	●	B	C	D	____
25.	A	●	C	D	____

Examination Outline Cross-Reference: EPE 007 G2.1.9 – Reactor Trip: Ability to direct licensed personnel activities inside the control room (SRO Only)	Level Tier Group # K/A # Rating	SRO 1 1 EPE 007 G2.1.9 4.5

Question 01

Unit 1 trips due to a spurious SI actuation.

Current RCS average temperature is 541°F and rising slowly.

In accordance with step 6, CHECK RCS Temperature, of EOP E-0, Reactor Trip or Safety Injection, the Shift Foreman directs the operator to stabilize RCS temperature at 541°F.

What is the reason the Shift Foreman directed stabilizing RCS temperature at its current value?

A. To ensure steam dumps are operating properly.

B. Maximize available subcooling for SI termination.

C. Prevent unblocking P-12 and re-initiating the RCS cooldown.

D. To extend the time to pressurizer overfill prior to SI termination.

Proposed Answer: D. To extend the time to pressurizer overfill prior to SI termination

Explanation:
The SRO knowledge addressing the KA is to understand *why* the action directed by the SRO is correct.
Step 6 is not an immediate action of EOP E-0 therefore, not RO knowledge The SRO must understand the plant event in progress and the proper procedural action to be taken depending on the RCS temperature and trend. If less than 547°F and rising, temperature is stablilzed at its current value. If above 547° and lowering the action is to continue to monitor RCS temperature. The reasons the action if above 547°F are to increase decay heat removal (if above and temperature is rising) or monitor for proper steam dump operation if above 547°F and temperature is lowering.

A. Incorrect. Plausible as this is the bases for the action if temperature is above no load and temperature is lowering.
B. Incorrect. Plausible because subcooling is the first criteria to be met when terminating subcooling and guarantees decay heat removal is adequate. Because the event is a spurious SI, its plausible to think this parameter should be maximized to allow SI termination.
C. .Incorrect. Plausible because if RCS temperature did continue to rise, P-12 could be unblocked, allowing operation of the steam dumps. The steam dumps would then control RCS temperature at no – load. No adverse cooldown would occur.
D. Correct. Once the cool down is stopped, RCS temperature should be stabilized at the CURRENT temperature using steam dumps (set to automatically control S/G pressure at current value). This will limit RCS pressure concerns, PZR overfill, and PZR PORV lifts later during a spurious SI or short-lived steam break (or similar event where EOP E-1.1 is

the appropriate recovery procedure). This is done even if temperature is stable at less than 547ºF. This aids in meeting the TCOA associated with Spurious SI.

Technical References: EOP E-0, LPE0-SI

References to be provided to applicants during exam: None

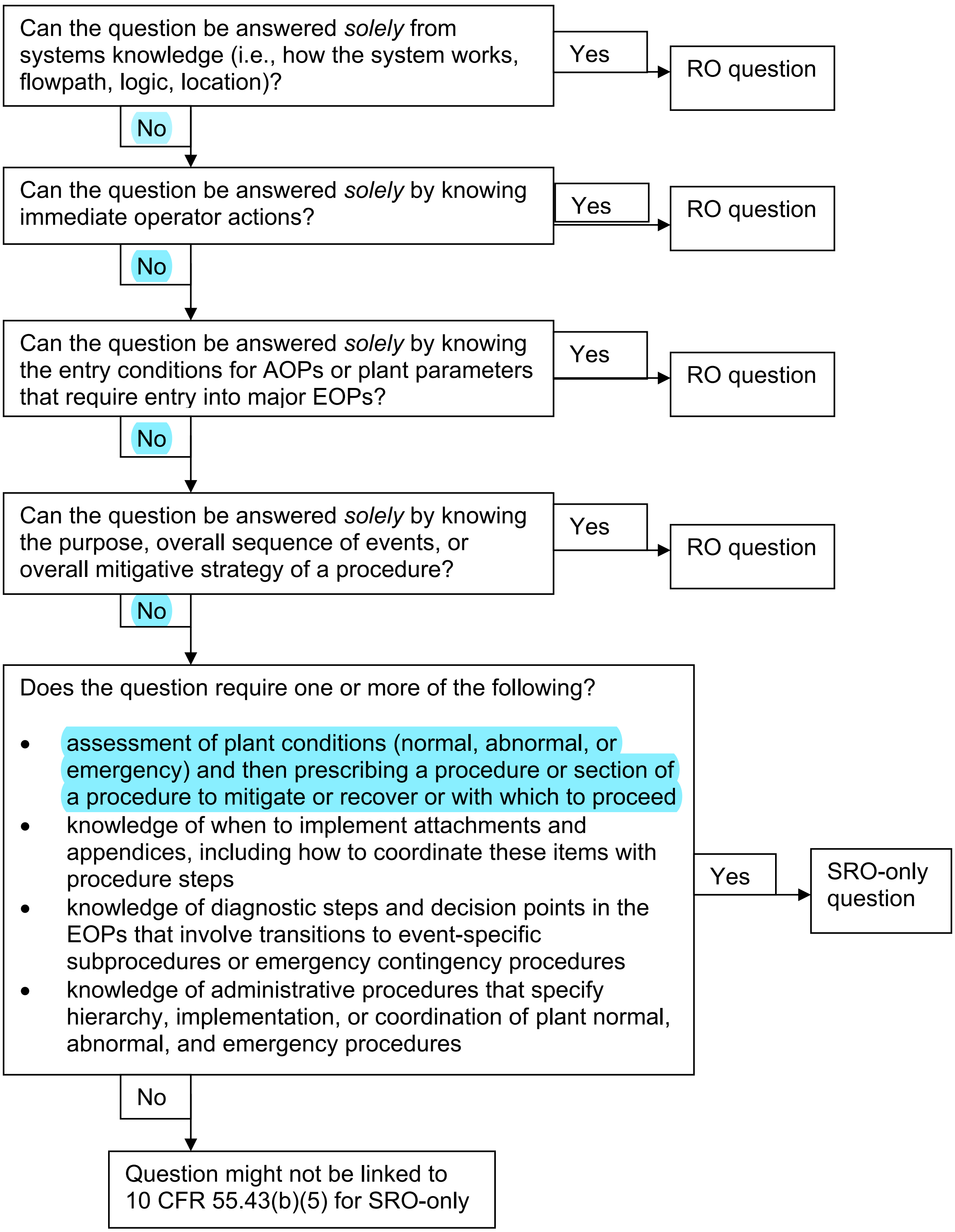
Learning Objective: Explain basis of emergency procedure steps in E-0 (7920A)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	N
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 3

Question 1

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
◊ **6. CHECK RCS Temperature:**	IF RCS Temperature is LESS THAN 547°F,
a. IF Any RCP is running,	THEN STABILIZE RCS Temperature using the following methods as needed:
THEN RCS T_{AVG} - STABLE OR LOWERING TO 547°F	1) Control Steam Dump Valves
OR	2) Control total feed flow while maintaining secondary heat sink.
b. IF NO RCPs are running,	3) IF Cooldown continues,
THEN RCS T_{COLD} - STABLE OR LOWERING TO 547°F	THEN Perform the following:
	a) Close ALL MSIVs and MSIV Bypass Vlvs. REFER TO Appendix L for local operation if necessary.
	b) Adjust 10% Steam Dump Controllers as needed to stabilize RCS temperature.
	4) GO TO step 7 (page 6).
	IF RCS Temperature is GREATER THAN 547°F and rising,
	THEN Dump steam to condenser
	OR
	Use 10% Steam Dump
	AND
	Stabilize RCS Temperature at LESS THAN OR EQUAL TO 547°F.

E-0 Diagnostics Steps

E-0 Diagnostic Steps & Basis
Obj 8, 10, 11, 17, 21,22

Review EOP E-0 diagnostic action steps and emphasize the following items:

- Action/Expected Response and Response Not Obtained basis
- Procedure transitions
- Other discussion items in the list below

Action	Basis and Discussion
Check RCS Temperature stable or lowering to 547°F. Make sure that you use the proper temperature based on RCP status. This step is a continuous action. A box around the entire step means it is a continuous action. These must be assigned to the CO or the BOPCO. When assigned a continuous action, write it down and refer to your list periodically. Closely monitor continuous actions and precisely control parameters in their proper bands. *Step 6*	• If cool down is the concern, steam dumps are ensured shut and AFW minimized (while maintaining 435 gpm until 15% [25%] to stop or minimize the cool down). – MSIVs are closed if cool down persists (App L if needed – local). – Note that excessive cool down cannot always be avoided (i.e., steam breaks, large LOCAs, etc.). • Once the cool down is stopped, RCS temperature should be stabilized at the **CURRENT** temperature using steam dumps (set to automatically control S/G pressure at current value). This will limit RCS pressure concerns, PZR overfill, and PZR PORV lifts later during a spurious SI or short-lived steam break (or similar event where E-1.1 is the appropriate recovery procedure). This is done even if temperature is stable at < 547ºF. • If temperature is higher than NOT, then steam dumps are used to stabilize temperature at NOT. • This step verifies that the steam dumps are controlling RCS temperature. The strategy to meet later TCOAs is to hand out this Page of procedure to board operator.

Continued on next page

Examination Outline Cross-Reference	Level	SRO
	Tier #	1
APE 015 - AA2.10 Ability to determine and interpret the following as they apply to Reactor Coolant Pump Malfunctions: loss of cooling or seal injection	Group #	1
	K/A #	APE 015 AA2.10
	Rating	3.8

Question 02

GIVEN:

- The crew has isolated CCW to RCP 1-1 thermal barrier by entering Containment and closing RCP 1-1 Thermal Barrier CCW return valve, CCW-1-234, in accordance with OP AP-28, Reactor Coolant Pump Malfunction, Section E, Loss of CCW to a RCP or RCP High Temperature
- A rapid plant shutdown, at 5 MWe/min, to remove RCP 1-1 from service is in progress
- Reactor power is 28% and lowering

The seal injection filter clogs and seal injection lowers to 0 gpm on all RCPs.

The Shift Foreman should take action in accordance OP AP-28,

A. Section D, Loss of Seal Injection, to direct the Aux Building watch to immediately swap seal injection filters, if unsuccessful in 5 minutes, direct the operator to trip the reactor, trip RCP 1-1 only.

B. Section D, Loss of Seal Injection, to direct the Aux Building watch to immediately swap seal injection filters, if unsuccessful in 5 minutes, direct the operator to trip the reactor and trip all the RCPs.

C. Section F, Loss of Seal Inj and Thermal Barrier Cooling Water, to direct the operator to trip the reactor and trip RCP 1-1 only.

D. Section F, Loss of Seal Inj and Thermal Barrier Cooling Water, to direct the operator to trip the reactor and trip all the RCPs.

Proposed Answer: C. Section F, Loss of Seal Inj and Thermal Barrier Cooling Water, to direct the operator to trip the reactor and trip RCP 1-1 only

Explanation: SRO must assess the change in plant conditions and determine what procedure section now applies and correct action to take.

A. Incorrect. Seal injection is lost, however, RCP 1-1 has lost both seal injection and thermal barrier cooling. Plausible if the focus is on the recent loss of seal injection and not on the larger picture of the effect on RCP 1-1. 5 minutes is plausible as this is a limitation for RCP operation for a loss of CCW to RCP motor coolers.
B. Incorrect. Section D is not appropriate to address the condition of cooling for RCP 1-1. Loss of CCW to all RCPs would require a trip of all RCPs, but cooling flow is only lost to one RCP. An immediate trip of all RCPs is not appropriate and would remove forced flow (the preferred source) as the decay heat removal mechanism. Plausible if it is believed the loss of seal injection due to seal filter clogging resulted in a loss of cooling to all RCPs, not just the 1-1 RCP.
C. Correct. Per OP AP-28, caution in section D and per step 1 of section F, an immediate trip and trip of the AFFECTED RCP is required if seal injection and CCW thermal barrier cooling is lost. Only the affected RCP is triped.

D. Incorrect. First part is correct, section F is the proper procedure section. Only 1 RCP must be tripped. Plausible if the action to trip all RCPs due to loss of CCW motor coolers is assumed to apply for a loss of seal injection as well or its assumed all RCPs are tripped and not just the one RCP.

References to be provided to applicants during exam: None

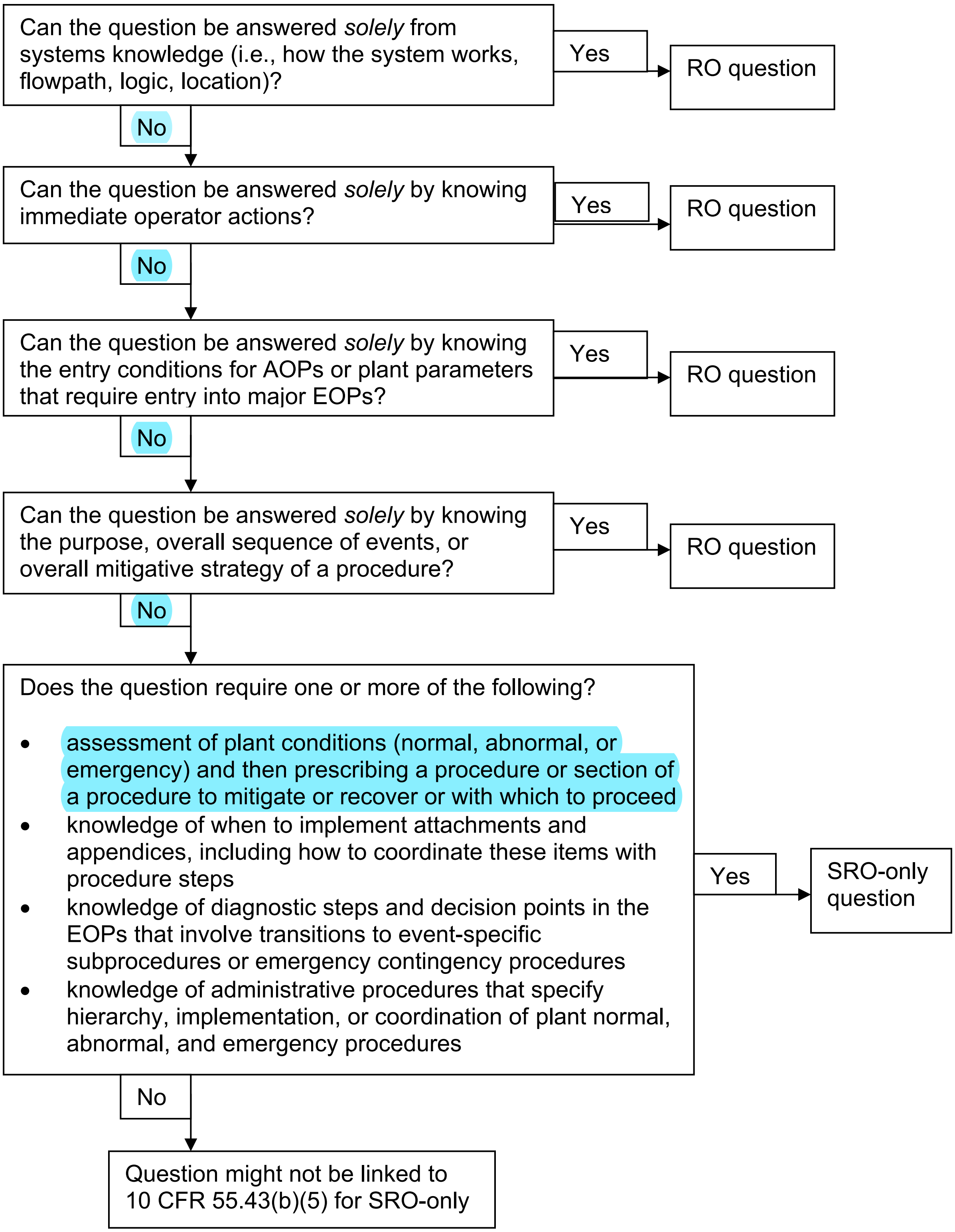
Technical Reference: OP AP-28, section D and F

Learning Objective: 7927 Given initial conditions and assumptions, determine if a reactor trip or safety injection actuation is required

Question Source:	Bank #86 L091C NRC exam	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC DCPP Exam 03/2012	Y
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	
Difficulty: 3		

Question 2

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Examination Outline Cross-Reference	Level	SRO
APE 003 G2.4.8 RCP: Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	**Tier #**	2
	Group #	1
	K/A #	APE003 G2.4.8
	Rating	4.5

Question SRO 11 (86) PARENT QUESTION

GIVEN:
- The crew has isolated CCW to RCP 1-1 thermal barrier by entering Containment and closing CCW-1- 234 in accordance with OP AP-28, Reactor Coolant Pump Malfunction, Section E, Loss of CCW to a RCP or RCP High Temperature
- A rapid plant shutdown to remove RCP 1-1 from service is in progress
- Reactor power is 28%

The seal injection filter clogs and seal injection lowers to 0 gpm on all RCPs.

In accordance with OP AP-28, which of the following actions should be taken by the Shift Foreman?

A. Direct the operator to trip the reactor, trip the 1-1 RCP and go to E-0, Reactor Trip or Safety Injection, while continuing with OP AP-28.

B. Direct the operator to trip the reactor, trip all the RCPs and go to E-0, Reactor Trip or Safety Injection, while continuing with OP AP-28.

C. Direct the Aux Building watch to immediately swap seal injection filters, if unsuccessful in 5 minutes, direct the operator to trip the 1-1 RCP and go to OP AP-25, Rapid Load Reduction or Shutdown, to perform a plant shutdown.

D. Direct the Aux Building watch to immediately swap seal injection filters, if unsuccessful in 5 minutes, direct the operator to trip the reactor, trip all the RCPs and go to E-0, Reactor Trip or Safety Injection.

Proposed Answer: A. Direct the operator to trip the reactor, trip the 1-1 RCP and go to E-0, Reactor Trip or Safety Injection, while continuing with OP AP-28.

Explanation:

A. Correct. Per OP AP-28, caution in section D and per step 1 of section F, an immediate trip and trip of the AFFECTED RCP is required if seal injection and CCW thermal barrier cooling is lost.
B. Incorrect. Loss of CCW to all RCPs would require a trip of all RCPs, but cooling flow is only lost to one RCP. A trip of all RCPs is not appropriate and would remove forced flow (the preferred source) as the decay heat removal mechanism.
C. Incorrect. Regardless of the power level, if an RCP must be stopped, the reactor is tripped first.
D. Incorrect. Only 1 RCP must be tripped and it must be immediately. 5 minutes is the time to restore CCW cooling to RCP lube oil coolers.

Technical References: OP AP-28

References to be provided to applicants during exam: None

Learning Objective: 7927 Given initial conditions and assumptions, determine if a reactor trip or safety injection actuation is required

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.	

Editorial - Added "IAW OP AP-28" to question.

SECTION D: Loss of Seal Injection

5. SYMPTOMS OR ENTRY CONDITIONS

5.1 Annunciator Alarms:

- RCP Seal Water Inj Flo Lo (PK05-01, 02, 03, 04)
- Charging Pp OC Trip, Stator Temp PPC (PK04-16, PK04-17, PK04-18)

5.2 Plant Indications:

- Trip light(s) ON at the charging pump control switch(es)
- Higher seal inlet temperature
- RCP seal injection flow indicating 0.0 gpm
- Higher radial bearing temperature

5.3 Automatic Actions:

5.3.1 High thermal barrier CCW return flow will cause auto closure of FCV-357, Thermal Barrier CCW Return Isolation Valve which isolates CCW return from ALL RCPs.

5.3.2 High temperature will cause actuation of RCP Shutdown Seal (SDS), and corresponding changes in seal leakoff flow.

SECTION D: Loss of Seal Injection (Continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	CAUTION: With NO Seal Injection flow AND NO Thermal Barrier cooling the affected RCP must be secured IMMEDIATELY to prevent damage to the pump.
1. ENSURE RCPs Thermal Barrier Cooling In Service:	GO TO SECTION F, this procedure, page 37.
a. FCV-750, RCP Thermal Barrier Return IC Valve - OPEN	
b. FCV-357, RCP Thermal Barrier Return OC Valve - OPEN	
c. RCP #1 Seal Outlet temperatures - NORMAL	
d. RCP Radial Brg Outlet temperatures - NORMAL	
e. CCW Header C alarm (PK01-08) - OFF	
2. CHECK At Least One Charging Pump - RUNNING	Perform the following:
• CCP 1 • CCP 2 • CCP 3	a. Ensure Letdown Flow Valves CLOSED: • 8149A, Letdown Orifice Isolation Valve • 8149B, Letdown Orifice Isolation Valve • 8149C, Letdown Orifice Isolation Valve
	b. Start a Charging Pump. IF A Charging Pump can NOT be started, THEN GO TO step 5, this section AND IMPLEMENT OP AP-17, "Loss of Charging."

SECTION E: Loss of CCW to an RCP or RCP High Temperature

6. SYMPTOMS OR ENTRY CONDITIONS

6.1 Annunciator Alarms:

- RCP Thermal Barrier CCW Flo Hi (PK 01-08)
- RCP 1 or 3 L.O. Clr CCW Flo Lo (PK 01-08)
- RCP Thermal Barrier CCW Flo Lo (PK 01-08)
- CCW Hdr-C Flo Lo (PK 01-08)
- RCP 2 or 4 L.O. Clr CCW Flo Lo (PK 01-08)
- RCP Temp PPC (PK05-01, 02, 03, 04)

6.2 Plant Indications:

- PPC alarms on RCP bearing temperatures
- PPC alarm on RCP stator temperatures
- Containment ambient temperature high

6.3 Automatic Actions:

6.3.1 High thermal barrier CCW return flow will cause auto closure of FCV-357, Thermal Barrier CCW Return Isolation Valve which isolates CCW return from ALL RCPs.

6.3.2 High temperature will cause actuation of RCP Shutdown Seal (SDS), and corresponding changes in seal leakoff flow.

SECTION E: Loss of CCW to an RCP or RCP High Temperature (Continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. ENSURE Seal Injection Flow to ALL RCP(s) - GREATER THAN OR EQUAL TO 6 GPM per Pump	GO TO SECTION D, this procedure, page 25.
2. ENSURE CCW To All RCP Lube Oil Coolers In Service: a. Ensure CCW Valves - OPEN • FCV-355 • FCV-356 • FCV-749 • FCV-363 b. RCP L.O. Clr CCW Flow LO Alarm (PK01-08) - NOT IN c. RCP Temp Alarm (PK05-01, 02, 03, 04) - NOT IN	IF CCW Flow to RCP(s) CANNOT be restored to Lube Oil Coolers within 5 minutes, THEN While continuing with this procedure, trip the Reactor AND IMPLEMENT EOP E-0, "Reactor Trip or Safety Injection." AND Stop the affected RCP(s).
3. ENSURE RCP parameters within OPERATING LIMITS PER Attachment 1	Perform the following: a. While continuing with this procedure, trip the Reactor AND IMPLEMENT EOP E-0, "Reactor Trip or Safety Injection." b. Stop the affected RCP(s). c. Higher seal injection flow to affected RCP(s) NOT to exceed 13 gpm, as necessary. d. IF RCP 1 or RCP 2 is affected, THEN Close associated spray valve: • RCP 1 – PCV-455A • RCP 2 – PCV-455B

SECTION E: Loss of CCW to an RCP or RCP High Temperature (Continued)

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**

NOTE: RCP(s) should NOT be run for extended periods without Thermal Barrier CCW Flow.

CAUTION: If FCV-357 closed on high flow, do NOT attempt to open FCV-357 until condition causing the high flow is cleared.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. ENSURE CCW Flow To All RCP Thermal Barriers:	PERFORM the following:
• Check FCV-357 did NOT close on high flow	a. Ensure FCV-750 CLOSED.
• Ensure Thermal Barrier Return Valves FCV-750 AND FCV-357 - OPEN	b. Enter containment and PERFORM the following: 1) Check Thermal Barrier flow to RCPs • FI-84 RCP 1 • FI-94 RCP 2 • FI-91 RCP 3 • FI-87 RCP 4 2) ISOLATE Thermal Barrier CCW return valve closing as applicable: • CCW-234 RCP 1 • CCW-242 RCP 2 • CCW-251 RCP 3 • CCW-262 RCP 4
• Ensure RCP Thermal Barrier CCW Flow Lo Alarm (PK01-08) - NOT IN	
	c. Monitor containment sump for expected level rise.
	d. IMPLEMENT OP AP-1," Excessive Reactor Coolant System Leakage."

- THIS STEP CONTINUED ON NEXT PAGE -

SECTION E: Loss of CCW to an RCP or RCP High Temperature (Continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. ENSURE CCW Flow To All RCP Thermal Barriers (Continued)	
	e. IF Thermal barrier CCW flow can NOT be restored, THEN PERFORM the following: 1) Continue attempts to restore RCP CCW flow while continuing this procedure. 2) Consult with Operations Manager to discuss if unit is to be shut down AND affected RCP(s) stopped.
	f. IF Shutdown is desired, THEN PERFORM the following: 1) Notify EPOS of Shutdown 2) Shut down the unit PER OP AP-25, "Rapid Load Reduction." 3) Stop the affected RCP(s). 4) IF RCP 1 or RCP 2 is affected, THEN close associated spray valve.

NOTE: High RCP motor winding temperatures may be caused by bus voltage either too high OR too low.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. CHECK ALL RCP Temperatures listed in Attachment 1 - STABLE IN NORMAL OPERATING RANGE	Perform the following as necessary to restore affected temperature(s): • Raise affected cooler CCW Flow. • Raise Containment cooling. • Adjust bus voltage (contact EPOS).

SECTION F: Loss of Seal Inj and Thermal Barrier Cooling Water

7. SYMPTOMS OR ENTRY CONDITIONS

7.1 Annunciator Alarms:

• RCP Thermal Barrier CCW Flo Hi	(PK 01-08)
• RCP 1 or 3 L.O. Clr CCW Flo Lo	(PK 01-08)
• RCP Thermal Barrier CCW Flo Lo	(PK 01-08)
• CCW Hdr-C Flo Lo	(PK 01-08)
• RCP 2 or 4 L.O. Clr CCW Flo Lo	(PK 01-08)
• RCP Seal Leakoff Flo Hi	(PK05-01, 02, 03, 04)
• RCP Seal Water Inj Flo Lo	(PK05-01, 02, 03, 04)
• RCP Temp PPC	(PK05-01, 02, 03, 04)
• RCP Pp Radial Brg Temp Hi	(PK05-01, 02, 03, 04)
• RCP Seal Leakoff Flo Lo (after Shutdown Seal actuation)	(PK05-01, 02, 03, 04)

7.2 Plant Indications:

- PPC alarms on RCP bearing temperatures.
- PPC alarm on RCP motor winding temperatures.
- Possible rise in RCP vibration.
- Elevated #1 seal leakoff flow or temperature
- Lowered #1 seal leakoff flow or temperature due to Shutdown Seal actuation
- Elevated radial bearing temperature.

7.3 Automatic Actions:

7.3.1 High thermal barrier CCW return flow will cause auto closure of FCV-357, Thermal Barrier CCW Return Isolation Valve which isolates CCW return from <u>ALL</u> RCPs.

7.3.2 High temperature will cause actuation of RCP Shutdown Seal, and corresponding changes in seal leakoff flow.

SECTION F: Loss of Seal Inj and Thermal Barrier Cooling Water (Continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. While continuing this procedure, TRIP the Reactor AND IMPLEMENT EOP E-0, "Reactor Trip or Safety Injection"	
2. TRIP affected RCP(s)	
3. CLOSE CVCS-8100, RCP #1 Seal Return Isol Vlv OC	Perform one of the following: • Close CVCS-8112, RCP #1 Seal Return Isol Vlv IC OR • Locally close CVCS-8100, RCP #1 Seal Return Isol Vlv OC (Pen Area 100-ft el. 103°) OR • Close CVCS-8396A, Seal Return Filter Inlet Isol (Filter Gallery)
4. CLOSE the Seal Injection Isolation Valve to affected RCP(s) • CVCS-8369A, RCP 1 Seal Water Injection Valve • CVCS-8369B, RCP 2 Seal Water Injection Valve • CVCS-8369C, RCP 3 Seal Water Injection Valve • CVCS-8369D, RCP 4 Seal Water Injection Valve	
5. ENSURE CCW-FCV-357, Thermal Barrier Clr CCW Return Isolation Valve - CLOSED	Manually close FCV-357, CCW Thermal Barrier Return Isolation Valve.

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	1
APE 026 G2.2.37 Loss of Component Cooling Water: Ability to determine operability or availability of safety-related equipment. (SRO Only)	**Group #**	1
	K/A #	APE 026 G2.2.37
	Rating	4.6

Question 03

Unit 1 is at 100% power. CFCU 1-1 is cleared with CCW to the CFCU isolated.

A leak develops in the coils of CFCU 1-2. CCW is isolated to CFCU 1-2 and the CFCU is declared inoperable.

At this time with CFCU 1-1 and 1-2 inoperable, action(s) per Technical Specifications is/are:

NOTE:
- LCO 3.6.6 - The containment fan cooling unit (CFCU) system and two containment spray trains shall be OPERABLE
- LCO 3.7.7 - Two vital CCW loops shall be OPERABLE

A. NOT required for either LCO 3.6.6 or LCO 3.7.7

B. required for LCO 3.6.6 only.

C. required for LCO 3.7.7 only.

D. required for both LCO 3.6.6 and 3.7.7.

Proposed Answer: B. required for LCO 3.6.6 only

Explanation:
CFCU's 1 through 5 are powered from vital buses F-F-G-H-G respectively.
Train A of CCW cools CFCUs 3 & 4
Train B of CCW cools CFCUs 1 & 2 & 5
SRO knowledge of SR 3.7.7.1 NOTE: Isolation of CCW flow to individual components does not render the CCW System inoperable. LCO 3.6.6 requires that either a minimum of 4 CFCUs or 3 CFCUs OPERABLE if powered from *different vital buses*.

A. Incorrect. Action is required. Both trains of CCW remain OPERABLE. Per SR 3.7.7.1 Note, isolation of CCW flow to individual components does not render the CCW system inoperable. However, two CFCUs are inoperable and while there are 3 remaining, they are not on different vital buses, therefore, LCO 3.6.6 is not met. Plausible because 3 CFCUs remain and if they were on different vital buses or if power supply knowledge is not accurate, this would be correct.
B. Correct. LCO 3.6.6 is not met. The remaining CFCU's are powered from bus G and H. This does not satisfy the LCO that if 3 are left, they must be on different 4 kv buses..
C. Incorrect. LCO 3.7.7 is still met. Isolating CCW to a component (per the note) does not make the system inoperable. Plausible because both CFCUs are on the same CCW header (B).
D. Incorrect. LCO 3.7.7 is still met. Isolating CCW to a component does not make the system inoperable. Plausible because both are on the B CCW header and LCO 3.6.6 is not met

Technical References: OVID 106714-4 & 106714-9 (CCW), LCO 3.6.6, LCO 3.7.7, B3.6.6

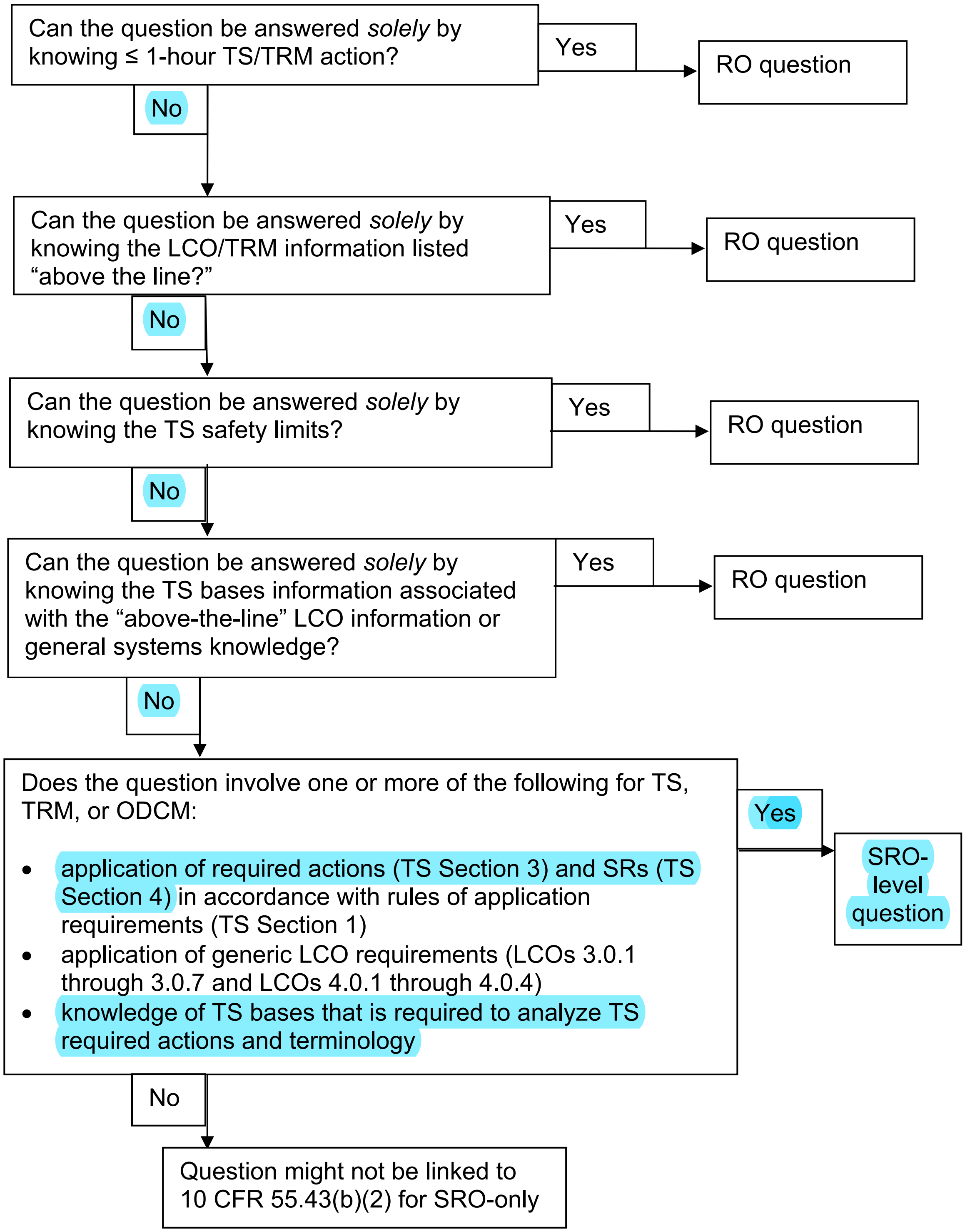
References to be provided to applicants during exam: None

Learning Objective: 9697F and G Apply TS 3.6 (3.7) Technical Specification LCOs

	Bank #	
Question Source: (note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	N
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.2	
Difficulty: 3		

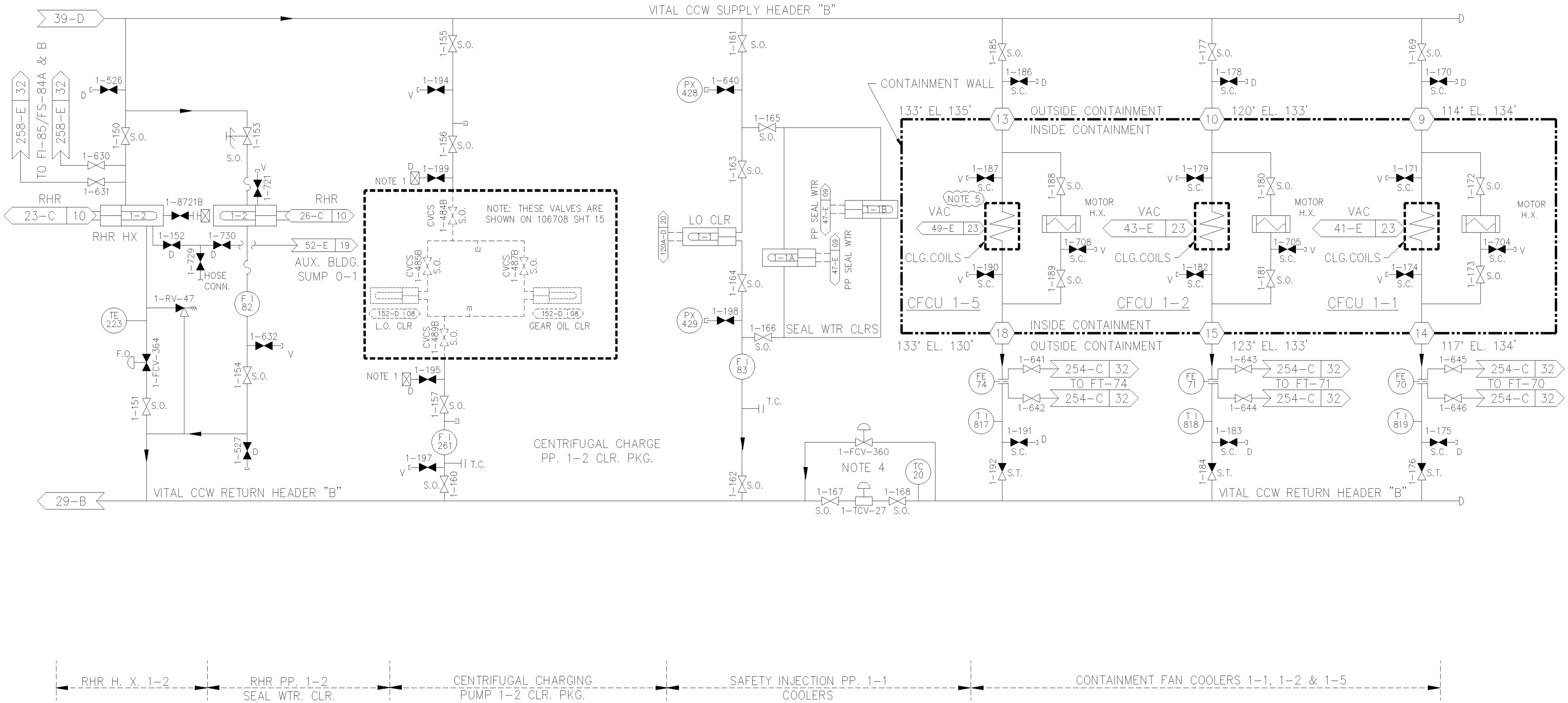
Question 3

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



c. *Facility Licensee Procedures Required to Obtain Authority for Design and Operating Changes in the Facility [10 CFR 55.43(b)(3)]*

Some examples of SRO-only examination items for this topic include the following:



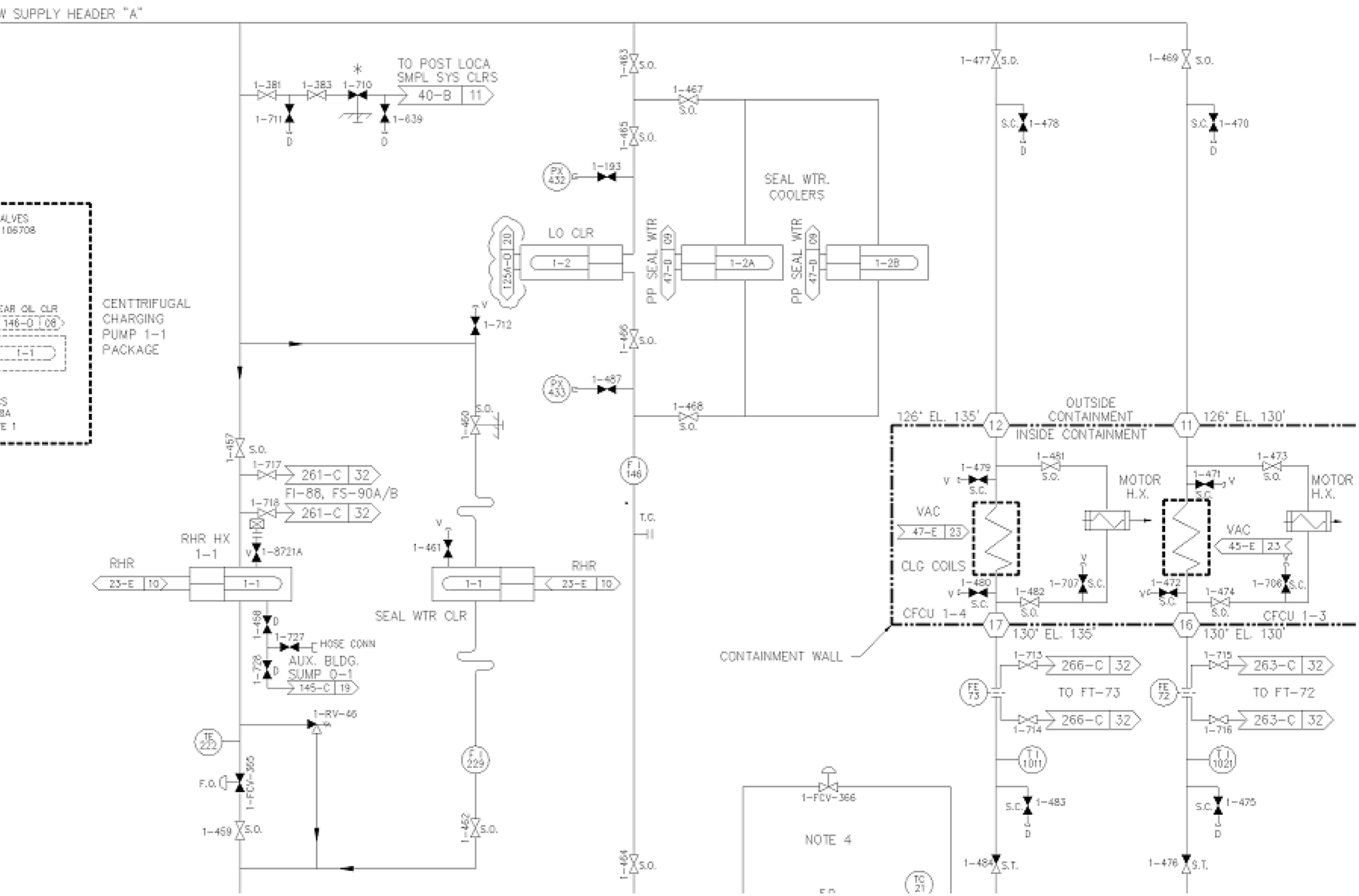
VITAL CCW HEADER "B" COMPONENTS

NOTES:

1. QUICK DISC. FOR B/U CLG WTR FROM FIREWTR SYS & DRN HOSES.
2. S.O.=SEALED OPEN
3. S.C.=SEALED CLOSE
4. TCV-27 & FCV-360 FAILED OPEN AND ABANDONED IN PLACE
5. RO's ARE INSTALLED AT THE COOLING COIL INLET NOZZLE FLANGES, TYP OF 6.

UNIT ONE

PG & E CO.	106714	REV. 75



W SUPPLY HEADER "A"
TO POST LOCA
SMPL SYS CLRS
40-B 11
SEAL WTR.
COOLERS
LO CLR
PP SEAL WTR 47-D 09
CENTTRIFUGAL
CHARGING
PUMP 1-1
PACKAGE
RHR HX
1-1
RHR
23-E 10
SEAL WTR CLR
261-C 32
FI-88, FS-90A/B
AUX. BLDG.
SUMP 0-1
145-C 19
HOSE CONN
OUTSIDE
CONTAINMENT
INSIDE CONTAINMENT
126" EL. 135'
126" EL. 130'
VAC
47-E 23
45-E 23
CLG COILS
MOTOR
H.X.
CFCU 1-4
CFCU 1-3
130" EL. 135'
130" EL. 130'
CONTAINMENT WALL
266-C 32
263-C 32
TO FT-73
TO FT-72
NOTE 4
1-FCV-366
1-FCV-365
1-RV-46

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray and Cooling Systems

LCO 3.6.6 The containment fan cooling unit (CFCU) system and two containment spray trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status. OR A.2 Restore containment spray train to OPERABLE status	72 hours ---------NOTE------------ For planned maintenance or inspections, the Completion Time is 72 hours. The Completion Times of Required Action A.2 are for unplanned corrective maintenance or inspections. ------------------------------ 14 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2. -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------- Be in MODE 4.	6 hours 54 hours
C. One required CFCU system inoperable such that a minimum of two CFCUs remain OPERABLE.	C.1 Restore required CFCU system to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One required containment spray train inoperable and one required CFCU system inoperable such that a minimum of two CFCUs remain OPERABLE.	D.1	Restore one required containment spray system to OPERABLE status,	72 hours
	<u>OR</u>		
	D.2	Restore one CFCU system to OPERABLE status such that four CFCUs or three CFCUs, each supplied by a different vital bus, are OPERABLE.	72 hours
E. Required Action and associated Completion Time of Condition C or D not met.	E.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	E.2	-----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------ Be in MODE 4.	12 hours
F. Two containment spray trains inoperable. <u>OR</u> One containment spray train inoperable and two CFCU systems inoperable such that one or less CFCUs remain OPERABLE. <u>OR</u> One or less CFCUs OPERABLE.	F.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	-----------------------------NOTE--------------------------- Not required to be met for system vent flow paths opened under administrative control. --- Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.2	Operate each CFCU for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.3	Verify component cooling water flow rate to each required CFCU is ≥ 1650 gpm.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.4	Verify containment spray locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.5	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6.6	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.7	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.8	Verify each CFCU starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.9	Verify each spray nozzle is unobstructed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.10	Verify each CFCU starts on low speed.	In accordance with the Surveillance Frequency Control Program

BASES

APPLICABLE SAFETY ANALYSES (continued)	sequenced loading of equipment, containment spray pump startup, and spray line filling (Ref. 1). The CFCUs performance for post accident conditions is given in Reference 1. The result of the analysis is that each train (two CFCUs) combined with one train of containment spray can provide 100% of the required peak cooling capacity during the post accident condition. The modeled Containment Cooling System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-High pressure setpoint to achieving full Containment Cooling System air and safety grade cooling water flow. The Containment Cooling System total response time includes signal delay, DG startup (for loss of offsite power), and component cooling water pump startup times. The Containment Spray System and the Containment Cooling System satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).
LCO	During a DBA LOCA, a minimum of two CFCUs and one containment spray train are required to maintain the containment peak pressure and temperature below the design limits (Ref. 1). Additionally, one containment spray train is also required to remove radioactive iodine and particulates from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains and the CFCU system consisting of four CFCUs or three CFCUs each supplied by a different Class 1E bus must be OPERABLE. Therefore, in the event of an accident, at least one train of containment spray and two CFCUs operate, assuming the worst case single active failure occurs. Each Containment Spray train typically includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal. Upon actuation of the RWST Low-Low alarm, the containment spray pumps are secured. Containment spray is then supplied by an RHR pump taking suction from the containment sump for a total spray operation (injection and recirculation) of 6.25 hours after accident initiation. The 6.25 hours accounts for the containment spray pumps startup time after accident initiation and the momentary containment spray shutdown time during switchover of the containment spray function from the containment spray pumps during injection phase to containment spray function being provided by the RHR system during recirculation phase. Management of gas voids is important to Containment Spray System OPERABILITY. Each CFCU includes cooling coils, dampers, fans, instruments, and controls to ensure an OPERABLE flow path.

(continued)

3.7 PLANT SYSTEMS

3.7.7 Vital Component Cooling Water (CCW) System

LCO 3.7.7 Two vital CCW loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One vital CCW loop inoperable.	A.1 ------------NOTE-------------- Enter applicable Conditions and Required Actions of LCO 3.4.6, “RCS Loops - MODE 4,” for residual heat removal loops made inoperable by CCW. ----------------------------------- Restore vital CCW loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------- Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	----------------------------NOTE---------------------------- Isolation of CCW flow to individual components does not render the CCW System inoperable --- Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.7.2	Verify each CCW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.3	Verify each CCW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

Examination Outline Cross-Reference		**Level**	SRO
		Tier #	1
APE 027 AA2.07 Ability to determine and/or interpret the following as they apply to a Pressurizer Pressure Control System Malfunction: Charging Flow Indication (CFR: 43.5 / 45.13)		**Group #**	1
		K/A #	APE 027 AA2.07
		Rating	3.4

Question 04

Unit 1 is in MODE 5. ECCS CCP 1-1 is in service.

The crew is preparing to place CCP 1-3 in service in accordance with OP B-1A:V, CVCS - Transfer Charging Pumps.

In accordance with LCO 3.4.12, LTOP System, Charging flow from CCP 1-3 should be minimized by ____________________1.__________________________.

According to the bases for LCO 3.4.12, it is acceptable to have more than one charging pump running during the pump swap to CCP 1-3 because credit is taken for__________2.__________ to limit the pressure increase during the pump transfer.

A. 1. aligning the LTOP orifice to CCP 1-3
   2. LTOP actuation opening all 3 PORVs

B. 1. aligning the LTOP orifice to CCP 1-3
   2. operator attentiveness and capability

C. 1. significantly throttling FCV-128, Charging Flow Control valve
   2. LTOP actuation opening all 3 PORVs

D. 1. significantly throttling FCV-128, Charging Flow Control valve
   2. operator attentiveness and capability

Proposed Answer: B. 1. aligning the LTOP orifice to CCP 1-3
2. operator attentiveness and capability

Explanation:
Question focuses on: 1 - the action taken to limit a potential pressure control malfunction during a period of plant operation when it is vulnerable to a pressure control malfunction occurring (two charging pumps running and LTOP in service) and 2 – why, according to the bases in Technical Specifications, its acceptable to be in a configuration of two charging pumps running for a short period of time while vulnerable to a pressure event occurring. (SRO knowledge, bases of TS)

A. Incorrect. First action is correct. The second part, reason is plausible – many exceptions in Tech Specs are allowed due to the small chance (ie of a LOCA) during the condition existing. All PORVs is plausible, during normal operation, all have the same setpoint and lift during an overpressure event. Also, if the bases for the LCO is not known, one could assume the over pressure transient causes pressure to quickly rise (especially if solid) and cause all PORVs to operate. A failure of equipment to actuate forms the bases for other specifications, such as the CFCU's which assumes a bus is lost and the LCO ensures that by requiring 3 CFCU's, two will be available to respond to an accident.

B. Correct. Per the procedure, (and as required by LCO 3.4.12) CCP 1-3 must be aligned to the LTOP orifice when in service. The bases for having more than one charging pump running states: Additionally, ECCS CCPs in excess of the above limitations can be momentarily capable of injection into the RCS for swapping of inservice ECCS CCPs. This condition is acceptable based on the operator's attentiveness to RCS pressure during the pump switch over and the capability of the operator to limit a pressure increase.
C. Incorrect. Action is incorrect. Plausible as this is the action if transferring from CCP 1-3 to an ECCS CCP. As stated in OP B-1A:V, if transferring to an ECCS CCP, charging is "substantially" throttled. Second part also incorrect. Additionally, LCO 3.4.12 bases states charging/letdown flow mismatch is a mass input type transient capable of overpressurizing the RCS. Second part is incorrect, it is allowed by taking credit for operator attentiveness and ability to control pressure.
D. Incorrect. Action is incorrect. Plausible as stated in OP B-1A:V, if transferring to an ECCS CCP, charging is "substantially" throttled. Additionally, LCO 3.4.12 bases states charging/letdown flow mismatch is a mass input type transient capable of overpressurizing the RCS.. Second part is correct.

Technical References: OP B-1A:V, TS bases B3.4.12

References to be provided to applicants during exam: None

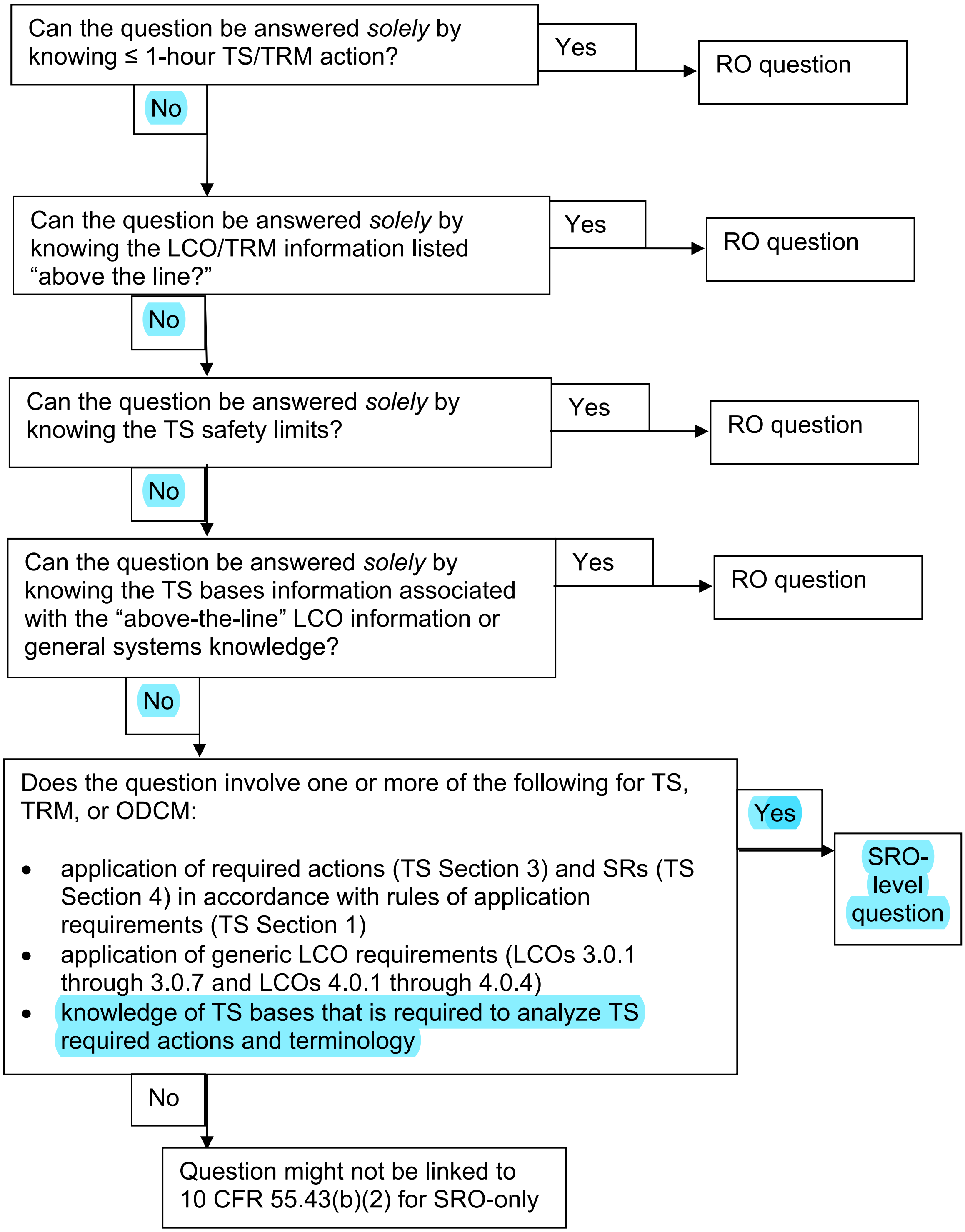
Learning Objective: 96794D, Apply TS 3.4 Bases

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	N
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.5	

Difficulty: 2.5

Question 4

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



c. *Facility Licensee Procedures Required to Obtain Authority for Design and Operating Changes in the Facility [10 CFR 55.43(b)(3)]*

Some examples of SRO-only examination items for this topic include the following:

5. PRECAUTIONS AND LIMITATIONS

5.1 Review the following technical specification/ECG items:

- ECG 8.4, Reactivity Control Systems - Flow Paths - Operating
- ECG 8.5, Reactivity Control Systems - Flow Path - Shutdown
- ECG 8.6, Reactivity Control Systems - Charging Pumps - Shutdown
- ECG 8.7, Reactivity Control Systems - Charging Pumps - Operating
- TS 3.5.2, ECCS - Operating
- TS 3.5.3, ECCS - Shutdown
- TS 3.4.12, Low Temperature Overpressure Protection System

5.2 Transferring charging pumps during solid plant operation is NOT recommended. This is due to the potential for rapid RCS pressure changes, which could have adverse effects on both RHR System operation and RCP operation.

5.3 The following guidance applies for swapping of CCPs while in Mode 4, 5, or 6 whenever LTOP is required:

5.3.1 Two charging pumps may be made capable of injecting for up to one hour for pump swap operation per TS 3.4.12.

5.3.2 Two CCPs may be run simultaneously during a swap but that should be minimized.

5.3.3 CCP 1-3 is to have its LTOP orifice inservice when required. If an ECCS CCP is the oncoming pump, flow MUST be throttled substantially to avoid a high flow mismatch condition.

5.4 Reactivity changes may result from placing idle portions of the CVCS in service due to differences in boron concentration in the piping. Use of the appropriate Excel spreadsheet on the Reactor Engineering webpage is recommended for determining net reactivity changes when changing the in-service charging pump. Perform a reactivity brief if required by the Reactivity Management Program.[T35167]

5.5 Avoid unnecessarily swapping charging pumps at low RCS pressure (less than 1000 psig). At low pressure, transients can occur on the pump seal packages when swapping charging pumps, leading to erratic seal return flow.[Ref 7.6.2]

5.6 If challenges occur during charging pump swaps, Pressurizer level should not be allowed to lower for the sole purpose of maintaining seal injection flow rates.[Ref 7.6.3]

5.7 Swapping the seal injection filters at the same time as the charging pumps can result in challenges to maintaining Pressurizer level and Seal Injection flow rates.[Ref 7.6.3]

BASES

BACKGROUND (continued)

The pressure relief capacity requires either two redundant RCS relief valves or a depressurized RCS and an RCS vent of sufficient size. One RCS relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

The pressurizer has three Power Operated Relief Valves. Two of the three are classified as PG&E Design Class I and are designated as “Class I” for LTOP pressure protection. All the PORVs are air operated. These two PG&E Design Class I PORVs have a nitrogen gas backup to the non-PG&E Design Class I air supply. Either air supply or nitrogen supply will satisfy PORV operability requirements.

The three PORVs are the same design. The PORV that is designated as ”non-Class I” may be used, when instrument air is available, to control RCS pressure similarly to the Class I PORVs although the non-Class I PORV does not receive an automatic open signal like the LTOP designated valves. Therefore, because no credit is taken for its operation for LTOP, continued operation with the non-Class I PORV unavailable for RCS pressure control is allowed as long as the associated block valve or non-Class I PORV can be closed to maintain the RCS pressure boundary.

In MODE 4 with the RHR loops in operation and in MODES 5 and 6, the operating RHR loop, connected to the RCS, can provide pressure relief capability through the RHR suction line relief valve. This capacity for RCS pressure relief is not assumed in the PTLR LTOP considerations and analyses and is not included in the LCO, ACTIONS, or Surveillances.

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the SI actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one ECCS CCP for makeup in the event of loss of inventory, then RHR pumps can be made available through manual actions.

Additionally, ECCS CCPs in excess of the above limitations can be momentarily capable of injection into the RCS for swapping of inservice ECCS CCPs. This condition is acceptable based on the operator's attentiveness to RCS pressure during the pump switch over and the capability of the operator to limit a pressure increase.

The LTOP System for pressure relief consists of two Class I PORVs with reduced lift settings or a depressurized RCS and an RCS vent of sufficient size. Two RCS Class I PORVs are required for redundancy. One RCS Class I PORV has adequate relieving capability to prevent overpressurization from the allowable coolant input capability.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, and in MODE 4 with RCS cold leg temperatures exceeding LTOP arming temperature specified in the PTLR, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At or below the arming temperature specified in the PTLR, overpressure prevention falls to two OPERABLE RCS Class I PORVs or to a depressurized RCS and a sufficiently sized RCS vent. Each of these means has a limited overpressure relief capability.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using the RCS relief valve method or the depressurized and vented RCS condition.

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 4 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Type Transients

a. Inadvertent safety injection;

b. Charging/letdown flow mismatch;

c. Accumulator discharge.

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of RHR cooling; or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

(continued)

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	1
EPE 038 G2.3.11 – SGTR: Ability to control radiation releases.	**Group #**	1
	K/A #	EPE 038 G2.3.11
	Rating	4.3

Question 05

The crew is preparing to transition out of EOP E-3, Steam Generator Tube Rupture.

According to the Westinghouse background document for a steam generator tube rupture, EOP E-3.1, Post-SGTR Cooldown Using Backfill, is the preferred recovery method because it:

A. minimizes radiological release and will prevent boron dilution of the RCS.

B. minimizes radiological release and will facilitate processing of contaminated primary coolant.

C. is the fastest of the recovery methods and will prevent boron dilution of the RCS.

D. is the fastest of the recovery methods and will facilitate processing of contaminated primary coolant.

Proposed Answer: B. minimizes radiological release and will facilitate processing of contaminated primary coolant

Explanation:

A. Incorrect. EOP E-3.1 will minimize radiological release, but a downside is backfill into the RCS could cause boron dilution, not prevent it. Plausible as this is an advantage of EOP E-3.2
B. Correct. According to the EOP E-3 background document, EOP E-3.1 minimizes the radiological release and allows for processing of the primary coolant.
C. Incorrect. EOP E-3.3 is the fastest method. Plausible to think the “preferred” method would be the fastest. Second part is incorrect. this is an advantage of EOP E-3.2.
D. Incorrect. First part incorrect. Plausible the “preferred method would be the fastest. Second part is correct.

Technical References: Westinghouse EOP E-3 background document

References to be provided to applicants during exam: None

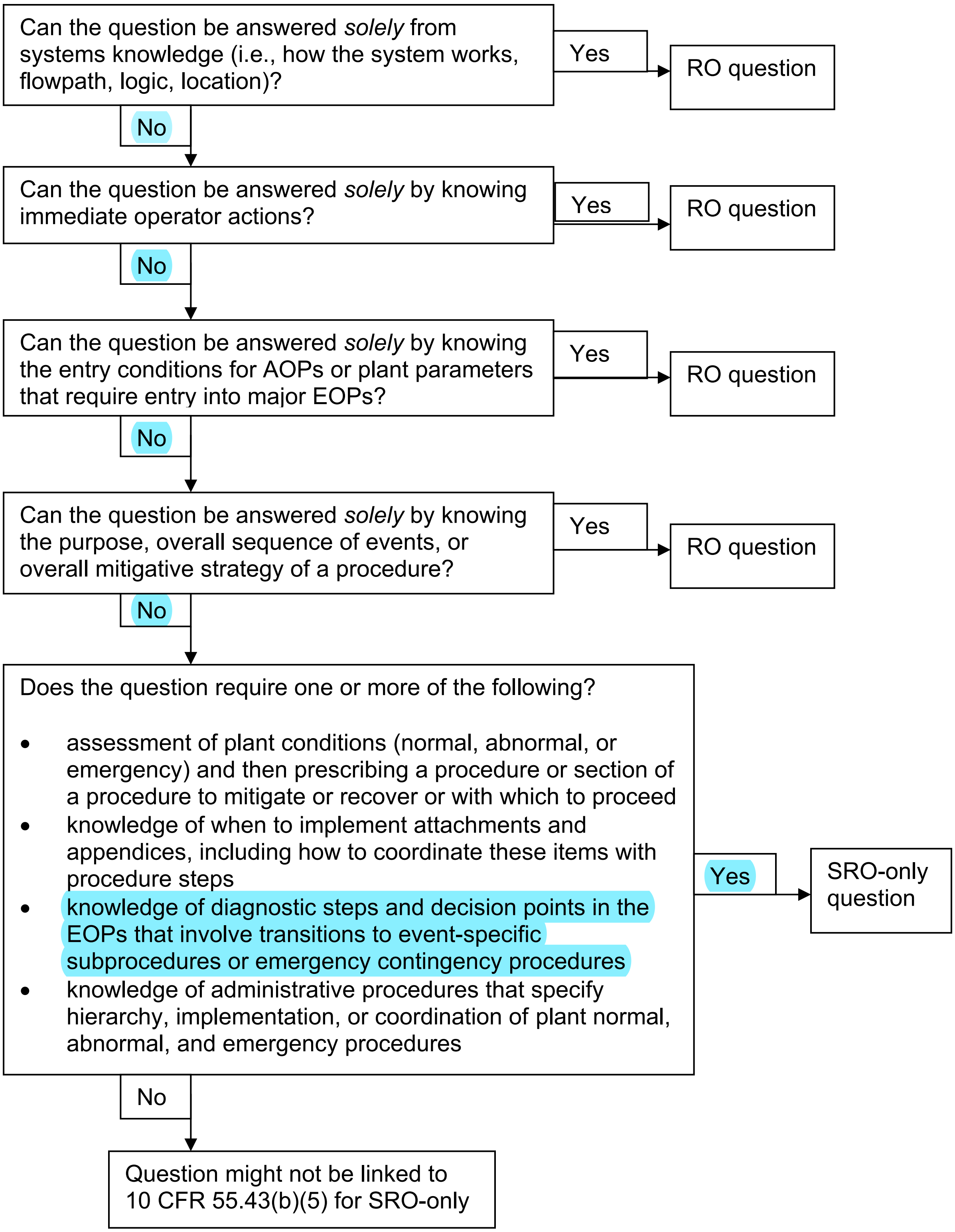
Learning Objective: 3552 - Given initial conditions, assumptions, and symptoms, determine the correct Emergency Operating Procedure to be used to mitigate an operational event

Question Source:	Bank #99 DCPP NRC exam L121	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC DCPP Exam 8/14	Y
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.4	

Difficulty: 2.0

Question 5

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Examination Outline Cross-Reference	Level	SRO
G2.3.11 - Ability to control radiation releases.	**Tier #**	3
	Group #	3
	K/A #	G2.3.11
	Rating	4.3

Question 99

Given the choice, why is E-3.1, Post-SGTR Cooldown Using Backfill, the preferred procedure for the Shift Foreman to enter from E-3, Steam Generator Tube Rupture?

A. To minimize radiological release and prevent boron dilution of the RCS.

B. To minimize radiological release and facilitates processing of contaminated primary coolant.

C. It is the fastest of the recovery methods and facilitates processing of contaminated primary coolant.

D. To prevent boron dilution of the RCS and facilitates processing of contaminated primary coolant.

Proposed Answer: B. To minimize radiological release and facilitates processing of contaminated primary coolant.

Explanation:
A incorrect – it will minimize radiological release, however, this process will be slow, particularly if no RCP is running. The fastest method is to dump steam (E-3.3).Boron dilution is a possibility to be considered when using this method since secondary water will be entering the RCS. It does facilitate processing

B correct – In general, post-SGTR cooldown using backfill is the preferred method since it <u>minimizes radiological releases and facilitates processing of contaminated primary coolant</u>

C incorrect – The fastest method is to dump steam, backfill will be slow, particularly if no RCP is running. It does facilitate processing of the contaminated primary coolant.

D incorrect – Boron dilution is a possibility to be considered when using this method since secondary water will be entering the RCS. It does facilitate processing.

Technical References: E-3 background

References to be provided to applicants during exam: None

Learning Objective: 3552 - Given initial conditions, assumptions, and symptoms, determine the correct Emergency Operating Procedure to be used to mitigate an operational event

Question Source:	Bank # DCPP NRC 02/2009 #97 (L061C)	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.4	

3.2 Key Utility Decision Points

The E-3 recovery guideline requires two key utility decisions which must be made on a plant and event specific basis. Information which can aid the utility is presented below.

o Select Post-SGTR Cooldown Method

When the instructions provided in E-3 are completed, primary-to-secondary leakage and radiological releases from the affected steam generator(s) should be stopped. However, the plant should be cooled and depressurized to cold shutdown conditions in order to make repairs and ensure no further radiological releases. This cooldown and depressurization is complicated by the ruptured steam generator which will act like a second pressurizer and inhibit RCS depressurization. Three alternate means of performing the post-SGTR cooldown have been developed and are presented in ES-3.1, POST-SGTR COOLDOWN USING BACKFILL, ES-3.2, POST-SGTR COOLDOWN USING BLOWDOWN, and ES-3.3, POST-SGTR COOLDOWN USING STEAM DUMP. The preferred method can be selected on a plant specific basis independent of the actual event. However, the actual method used depends on the available plant equipment and the evolution of the event.

In general, ES-3.1, POST-SGTR COOLDOWN USING BACKFILL, is the preferred method since it minimizes radiological releases and facilitates processing of contaminated primary coolant. However, the chemistry of the secondary side water should be considered with respect to adverse effects on primary system components and potential boron dilution prior to initiating backflow of secondary side fluid. The steam generator blowdown method, ES-3.2, can also minimize radiological releases. In addition, boron dilution and adverse secondary side water chemistry concerns are eliminated. However, the storage and processing capabilities of the blowdown system may be limited. Consequently, this method could spread contamination to secondary side components resulting in increased occupational exposure or offsite doses due to discharge from the blowdown system or contamination of the intact steam generator feedwater supply.

Both the backfill method and blowdown method are likely to proceed slowly since they may require repeated draining of the affected steam generators through small capacity lines. This is particularly true if no RCP is running. In that case, the water in the steam generator tube region may remain warmer than the rest of the RCS and flash to steam as pressure is decreased. In some cases, it may be desirable to establish RHR System cooling more quickly, e.g. to conserve feedwater supply or to minimize radiological releases from the intact steam generators. To provide a more rapid means of depressurizing the RCS and ruptured steam generators, a post-SGTR cooldown method using steam dump, ES-3.3, was developed. This is the fastest of the three methods. However, the radiological consequences must be considered particularly if steam dump to the condenser is unavailable. The NRC has indicated that controlled radiological releases from the affected steam generators should not exceed 10 CFR 20 limits. In addition, if water exists in the steamline, steam release may cause water hammer effects resulting in damage to secondary side equipment. Consequently, this method should not be used if water may exist in the main steamlines.

Although the three post-SGTR cooldown guidelines are presented as alternate methods, they are similar. Consequently, one could begin with the backfill method, continue with the blowdown method, and complete the recovery using steam dump, provided the limitations of each method are observed. Similarly, for multiple tube failures, one could execute combinations of the three methods at the same time. For example, one could depressurize one ruptured steam generator using blowdown and another using the backfill method.

These guidelines provide the flexibility necessary to cool down and depressurize the plant to cold shutdown conditions for a wide variety of SGTR scenarios and plant designs. Each utility must evaluate the three methods on a plant specific basis to establish a preferred post-SGTR cooldown method and prioritize the alternate methods. The actual recovery method must be determined on an event specific basis with consideration of the available equipment and evolution of the event. For example, normal letdown may be needed to complete recovery using the backfill method, ES-3.1, while steam generator blowdown is needed to implement ES-3.2. The guideline ES-3.3 should not be used if water may exist in the main steamlines or steam releases may lead to unacceptable radiological releases.

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	1
APE 056 AA2.53 Ability to determine and/or interpret the following as they apply to Loss of Offsite Power: Status of emergency bus under voltage	**Group #**	1
	K/A #	APE 056 AA2.53
	Rating	3.3

Question 06

Unit 1 is in MODE 4 aligned to Startup power.

A degraded voltage condition causes all 4 kV Vital Bus voltages to suddenly lower to approximately 0% of rated voltage.

1. The _____________ will automatically energize the Shed Loads relay on its respective 4 kV Vital Bus.
2. To restore decay heat removal, the Shift Foreman should go to:

NOTE:

FLUR – First Level Undervoltage Relay
SLUR – Second Level Undervoltage Relay

A. 1. FLURs
   2. OP AP-16, Malfunction of the RHR System

B. 1. FLURs
   2. OP AP-27, Loss of Vital 4 kV and/or 480 VAC Bus

C. 1. SLURs
   2. OP AP-16, Malfunction of the RHR System

D. 1. SLURs
   2. OP AP-27, Loss of Vital 4 kV and/or 480 VAC Bus

Proposed Answer: A. 1. FLURs
2. OP AP 16, Malfunction of the RHR System

Explanation:
FLUR actuate and transfer to the diesel in 4 seconds (response to a loss of power to the bus). SLUR transfer to diesel has a 20 second delay (response to a degraded bus condition) For a loss of a bus, the FLUR will actuate to initiate shed loads and energize the bus from the diesel. The procedure to restore decay heat removal will be OP AP-16 which will instruct the crew to start an RHR pump. OP AP-27 deals with restoring power to the bus, which is done by the actions of the relays.
NOTE: The SRO must decide which AOP will address the issue of the loss of decay heat removal. While entry conditions into either AOP may be met, the SRO must know which one is the priority and will address restoring RHR.

A. Correct. FLUR is actuated for lower voltage (ie loss of bus voltage). The diesels are started and the output breaker closes to energize the vital 4kV bus. However, the RHR pumps, although previously running, should not be restarted and the crew will go to OP AP 16. First step is to Verify RHR pump running – this means the crew will start the RHR pump.

B. Incorrect. First part is correct. FLUR is actuated for lower voltage (ie loss of bus voltage). Second part is incorrect. While the bus will be stripped by the Shed Loads Relay, the diesels will start and re-energize the bus (however, the RHR pumps are not restarted). OP AP-27 deals with re-energizing the bus, which occurs automatically. Plausible because other pumps, such as ASW, will be energized when the diesel energizes the bus.
C. Incorrect. SLUR will actuate and initiate a transfer to diesel for degraded voltage conditions, however, it is for a degraded bus condition – the FLUR is for loss of bus conditions and actuates without a time delay to begin the shed load process. The diesels are started and the output breaker closes to energize the vital 4kV bus. However, the RHR pumps, although previously running, will not be restarted and the crew will go to OP AP 16. First step is to Verify RHR pump running – this means the crew will start the RHR pump. Second part is correct.
D. Incorrect. First part is incorrect. Plausible because a degraded voltage condition will cause SLUR actuation. Second part incorrect but plausible if it's thought actions are necessary to energize the 4 kV buses and/or start an RHR pump in accordance with the loss of power procedure.

Technical References: OIM page J-5-1e & J-6-1, LJ-15, OP AP-16, OP AP-27

References to be provided to applicants during exam: None

Learning Objective: Given initial conditions, assumptions, and symptoms, determine the correct abnormal operating procedure to be used to mitigate an operational event. (3478)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	N
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 3

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)

Can the question be answered *solely* from systems knowledge (i.e., how the system works, flowpath, logic, location)?

Yes → RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No

Can the question be answered *solely* by knowing the entry conditions for AOPs or plant parameters that require entry into major EOPs?

Yes → RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No

Does the question require one or more of the following?

- assessment of plant conditions (normal, abnormal, or emergency) and then prescribing a procedure or section of a procedure to mitigate or recover or with which to proceed
- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific subprocedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, or coordination of plant normal, abnormal, and emergency procedures

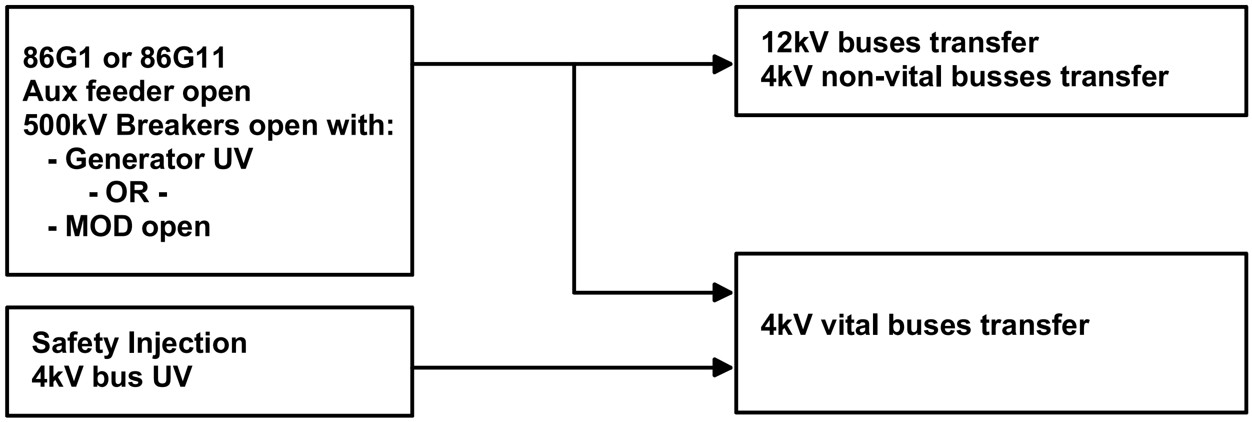
Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Auto Bus Transfers

Bus Transfer Signals and Actions



Equipment Loading Times

Equipment	Transfer to SU (no SI)	Transfer to DG (no SI)	Transfer to SU or DG (SI)	Bus	Train*
ASW 1 / 2	10	10	14	F/G	A/B
AFW 2 / 3	X	14	18/26	H/F	B/A
CCP 1 / 2	X	20	2	F/G	A/B
SI 1 / 2	X	X	6/2	F/H	A/B
RHR 1 / 2	X	X	6	G/H	B/A
CCW 1 / 2 / 3	X	5	10/10/14	F/G/H	A/B/A&B
CS 1 / 2	X	X	26 [1]	G/H	B/A
CFCU 1 / 2 / 3 / 4 / 5	25 [2]	25 [2]	22/18/18/22/22 [2]	F/F/G/H/G	A/A/B/A&B/B

Notes:

1. With a containment Hi-Hi signal only.
2. Starts in low speed only.

All times are in seconds.
X = Not started.

*OP1.DC38, Att 8.2, Pg 2

Ref: 458863
458864
458865
437600
19407
STP M-13B Series

4kV Buses F, G, And H Auto Transfers
(Continued)

Sequence of events for a "Transfer to Diesel" (startup power not available):

1. The diesel generator is started, and a Transfer to Diesel initiated, by:
   - 4KV bus undervoltage - First level (inverse time relays pickup at approximately 80% rated voltage)
     - Set for 0.8 seconds at 0 volts for diesel start
     - Set for 4 seconds at 0 volts for transfer to diesel
   - 4KV bus undervoltage - Second level (instantaneous relays pickup at approximately 90% rated voltage)
     - 10 second time delay for diesel start
     - 20 second time delay for transfer to diesel

   (An S.I. signal and loss of startup voltage starts the diesel generators, but does NOT initiate a transfer to diesel)
2. All motor breakers on associated bus are tripped open by the Shed Loads Relay.
3. When diesel generator is up to rated voltage the:
   - Auxiliary and startup feeder breakers are tripped open and locked out
   - Diesel output breaker closes after a 2 second time delay
4. Sequencing timers start selected loads on the bus (See J-6-1)

Auto "Transfer to Diesel" is blocked if the:

1. Auxiliary feeder breaker tripped on "Overcurrent"
2. Startup feeder breaker tripped on "Overcurrent"
3. Bus "Differential" action has occurred

The diesel generator engine will be tripped and locked out (Shutdown Relay actuated) by a:

1. Diesel "Generator Differential" relay trip.
2. Diesel engine low lube oil pressure while running.
3. Diesel Engine "Overspeed" occurs.
4. The "Emergency Shutdown" device is actuated (button outside the diesel generator room).
5. High jacket water temperature or overcrank trips are only active when in "Local" control and "Test" (manual) (low jacket water pressure for greater than 10 seconds).

The diesel generator breaker is tripped open if the:

1. "Shutdown Relay" trips (listed above).
2. Diesel "Generator Differential" relay trips.
3. Associated 4kV bus "Differential" relay trips.
4. Diesel generator "Reverse Power" relay trips.
5. Diesel generator "Loss of Field" relay trips.
6. Diesel generator breaker "Overcurrent" relay trips.

Notes:
* Trips are normally cutout (non-seismic relays)
See J-6-3b for setpoints
On any Diesel Generator starting signal, the Start Air Solenoids will be de-energized (closed) if the Engine cranks for >10 seconds and does not start.

Ref: 437580	437666
458863	437617
437625	T-MOD 60024240

4 kV Vital Bus Shed Loads Relay, Continued

Objective 10 Analyze automatic features and interlocks associated with the Electrical Power Transfer System.

- 4 kV Vital Bus Shed Loads Relay

Study Topics

- **4 kV Vital Bus Shed Loads Relay initiating signal logic described in the table below (refer to Figure EPT-21 on the next page).**

If either of the following initiating signals are received (Bus F described)	Then the shed loads relay will ...
• First level Undervoltage relays (FLURs) • (27HFT1) - inverse time, ≈ 70% • pickup volts, 4 sec time delay @ 0 volts AND • (27HFT2) - ≈ instantaneous, ≈ 70%	Actuate
• Second level Undervoltage relays (SLURs) • (27HFB3) - instantaneous, ≈ 91% AND • (27HFB4) - instantaneous, ≈ 91% AND • D/G output bkr is open (or paralleled with Aux) AND • 20 seconds have elapsed following SLUR actuation	

Note: SLURs are designed to protect against sustained degraded voltage conditions.
Note: The 4 second TD for the FLURs is designed to allow the bus to transfer to S/U prior to stripping the bus of its loads
Note: The 20 second TD for the SLUR is designed to allow the bus voltage to return to Normal following a transient before stripping bus loads.

Continued on next page

TITLE: Malfunction of the RHR System

1 AND 2

INFO ONLY

EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

1.1 This procedure covers the steps to be taken following malfunction of the Residual Heat Removal system due to a loss of flow or leakage while in Mode 4.

1.2 OP AP-24, "Shutdown LOCA," provides the action necessary for maintaining core cooling and protecting the reactor core in the event of an RCS leak while on Residual Heat Removal. This procedure may be implemented from OP AP-24 to provide diagnosis for and corrective actions for leaks occurring in systems other than the RCS that cause a loss of RCS inventory.

2. SYMPTOMS

2.1 Abnormal fluctuations in RHR flow on FI-970A & B and/or FI-971A & B.

2.2 Increasing level indication in the PRT.

2.3 Increasing level indication in CCW Surge Tank.

2.4 Possible Annunciator Alarms

2.4.1 CCW SYS SURGE TK LVL/MK-UP (PK01-07)

a. CCW Surge Tk Level Hi/Lo

2.4.2 RHR SYSTEM (PK02-16)

a. RHR Suction Valve Open

b. RHR Pp _____ Room Sump Lvl Hi

c. RHR Pp _____ Disch Press Hi

d. RHR Pp 2-1 and 2-2 Disch Flow Low

2.4.3 RHR PUMPS (PK02-17)

a. RHR Pps OC Trip

b. RHR Pp _____ Room Sump Pps run

2.4.4 HIGH RADIATION (PK11-21)

a. Contmt Area Mon High Rad RE-2

b. Process Mon Hi-Rad (RE-17A & B)

2.4.5 SUBCOOLING MARGIN LO/LO-LO (PK05-07)

a. Subcooled Mon Lo-Lo Margin

b. Subcooled Mon Lo Margin

2.4.6 PRT PRESS/LVL TEMP (PK05-25)

a. PZR Relief Tk Lvl Hi-Lo

2.4.7 CONTMT/TB SUMP 0-1 (PK15-01)

a. Containment Sump ______ Lvl Hi

b. Reactor Cavity Sump Lvl Hi

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. CHECK RHR Pump Status:	
a. Verify a RHR Pump running	a. GO TO Step 2. Observe CAUTION prior to Step 2.
b. Verify RHR flow indicated on FI-970/971	b. Verify that FCV-641A/B (as appropriate) is open and RHR pump motor current is ≤28 amps. Check RHR valve alignment and restore flow to vessel. IF Recirc flow cannot be verified, THEN Stop the running RHR pump(s) and check the RHR valve alignment. Restart the RHR pumps when the alignment has been corrected.
c. VERIFY CCW to the RHR Heat Exchangers	c. Implement OP AP-11.

****

CAUTION: To prevent RCP Seal damage, maintain RCP No. 1 seal Delta-P greater than 255 psid whenever an RCP is in service.

****

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. START An Alternate Means Of Heat Removal, if required:	
a. Check RHR not in operation	a. GO TO Step 3.
b. Check at least one RCP in operation	b. Implement Appendix B to Start an RCP.
c. Control RCS temperature using condenser steam dumps as necessary	c. Control RCS temperature by dumping steam to the atmosphere using S/G 10% atmospheric steam dumps.
3. CHECK For Indication of Leakage From the RHR System	
a. No indication of leakage from the RHR system (see Symptoms)	a. GO TO Step 4. Observe NOTE prior to Step 4.
b. Check RHR in Operation	b. RETURN TO Step 1.
c. RETURN TO Procedure and Step in Effect	

Loss of Vital 4 kV and/or 480 VAC Bus

05/20/19
Effective Date

QUALITY RELATED

1. SCOPE

1.1 This procedure covers the steps to be taken in the event of a loss of power to a vital 4 kV and/or 480 VAC bus.

1.2 This procedure should be used in MODES 1-4. If in MODE 5 or 6, OP AP SD-1, "Loss of AC Power," should be used.

2. SYMPTOMS OR ENTRY CONDITIONS

2.1 Possible Main Annunciator Alarms:

2.1.1 4KV BUS DIFF LOCKOUT (PK16-16, 17-16, or 18-16)

2.1.2 4KV BUS OR SU FDR UV (PK16-17, 17-17, or 18-17)

2.1.3 480V BUS FEEDER TOL OR OC TRIP (PK16-22, 17-22, or 18-22)

3. INSTRUCTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. CHECK ANY 4 kV OR 480 VAC vital bus - DEENERGIZED • White potential light - OFF	Return to procedure and step in effect.
2. CHECK DRPI - ENERGIZED • Rod position lights - LIT	a. Place rod control in MANUAL. b. IF ramp in progress, THEN place ramp on HOLD. c. IMPLEMENT AR PK03-21.
3. CHECK alternate pumps - RUNNING off any energized 4 kV buses	
a. Ensure ASW Pump running	a. IMPLEMENT OP AP-10 for loss of ASW.
b. Ensure 2 CCW Pumps running	b. IMPLEMENT OP AP-11 for CCW system malfunctions.
c. Check Charging Pump running	c. IMPLEMENT OP AP-17 Section A to restore charging and letdown.
4. CHECK ALL 480 VAC vital buses - ENERGIZED • White potential lights - LIT	a. IMPLEMENT OP AP-4, LOSS OF VITAL OR NON-VITAL INSTRUMENT AC, for any non-vital 120 VAC distribution panel that may have been deenergized. b. IMPLEMENT the following for the deenergized 480 VAC vital busses: • Section 6 for loss of 480 VAC Bus F • Section 7 for loss of 480 VAC Bus G • Section 8 for loss of 480 VAC Bus H
5. CHECK bus DEENERGIZED due to over-current or differential • Blue light - ON	Inspect bus switchgear and attempt to reenergize PER one of the following: • 4 kV - OP J-6A:I • 480 VAC - OP J-7B:I
6. CONTACT maintenance to investigate	

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**

7. **REFER TO the following technical specifications:**
   a. TS 3.1.7, Rod Position Indication
   b. TS 3.4.11, PORVs
   c. TS 3.5.2, ECCS - Operating
   d. TS 3.6.6, Containment Spray and Cooling Systems
   e. TS 3.7.5, Auxiliary Feedwater System
   f. TS 3.8.1, AC Sources - Operating
   g. TS 3.8.4, DC Sources - Operating
   h. TS 3.8.7, Inverters - Operating
   i. TS 3.8.9, Distribution Sys - Operating
   j. TS 3.6.3, Containment Isolation Valves
   k. TS 3.7.3, MFIVs, MFRVs, MFRV Bypass Valves, MFWP Turbine Stop Valves
   l. ECG 8.1, Charging Pump 1-3
   m. ECG 23.6, 480 VAC Class I Switchgear Ventilation System

8. **STOP any unloaded diesel generator**
   a. Place DG MODE SEL switch in "MANUAL"
   b. Momentarily turn start/stop switch to "STOP"

9. **CHECK ESF equipment OPERABILITY**
   a. AFW Pumps
   b. Containment Spray Pumps
   c. RHR Pumps
   d. SI Pumps
   e. ECCS Charging Pumps

10. **RETURN to procedure and step in effect**

- END -

4. SUPPLEMENTAL INSTRUCTIONS

4.1 Section 6, "Loss of 480 VAC Bus F Supplemental Actions"

4.2 Section 7, "Loss of 480 VAC Bus G Supplemental Actions"

4.3 Section 8, "Loss of 480 VAC Bus H Supplemental Actions"

5. ATTACHMENTS

None

6. Loss of 480 VAC Bus F Supplemental Actions

NOTE:	Reference Dwg. 437916 section 1C - PK18-22 Input 135.

6.1 IF DRPI was being supplied by normal power (52-1F-45), THEN perform the following:

6.1.1 Ensure Rod Control in manual AND REFER TO TS 3.1.7.

6.1.2 Place DRPI on backup power PER OP A-3:I.

6.2 With Block Valve 8000A deenergized, within one hour place PCV-474 hand switch to close. REFER TO TS 3.4.11.

6.3 Assign personnel to perform necessary Nuclear Operator actions, steps 6.7 - 6.11 (battery charger transfer is a 2-hour action.)

6.4 Ensure redundant equipment on other 480 VAC buses is running.

- Primary Makeup Water Pump 1-2
- Boric Acid Transfer Pump 1-2 aligned to blender
- FHB Exhaust Fan E-4 or E-6
- CFCUs 1-3, 1-4, and 1-5
- Aux Bldg to Exhaust Fan E-2

6.5 Ensure the following equipment is NOT on backup power supply.

- Battery 1-3 NOT on charger 1-31 (OP J-9:II)

6.6 Address the following issues as time permits.

6.6.1 FWP 1-1 turning gear deenergized.

6.6.2 Aux feedwater supply LCVs 115 and 113 require local operation.

6.6.3 Check status of vital instrument inverters.

NOTE:	Diesel crankcase exhausters are desired, but not required, for starting and normal operation of the diesel generator. Reference AR A0503292.

6.6.4 DG 1-3 may be running without crankcase exhausters - consider shutting it down (Aux Motor Contactor Panel 'A').

6.6.5 ASW Pump may be running without exhaust fan cooling (E-104) if 4 kV bus still energized. Investigate status of continuous chlorination.

6.6.6 Numerous valves fail as-is and will require local operation if necessary. See Fed From list for a complete listing (52-HF-10), including FCV-601 (REFER TO ECG 17.1).

6.6.7 Diesel Generator 1-2 Aux Motor Contactor Panel 'B' is lost.

- Consider installing air cross-tie hose per OP J-6B:II.
- Monitor lube oil temperature (≥ 90°F, TS SR 3.8.1.2 Basis).

CAUTION:	Diesel Generator margin assessment shall be completed at the earliest opportunity and if possible prior to transferring loads to their alternate vital bus to ensure the DG will not be overloaded during a loss of offsite power.

NOTE:	With the exception of pressurizer heaters, all loads capable of being aligned to a vital bus as an alternate source can be aligned to the vital bus simultaneously provided any ECCS pump powered from that bus is prevented from starting.

6.7 On any vital bus receiving additional load due to transfer of plant equipment to alternate source of power, perform diesel generator margin assessment PER OP O-13, "Transferring Equipment to/from Alternate Power Source." Reference SAPN 50350316.

6.8 IF DRPI was being supplied by normal power (52-1F-45),
THEN place DRPI on backup power PER OP A-3:I.

6.9 Transfer the following equipment to backup power as necessary.

- Battery 1-1 to charger 1-21 PER OP J-9:II.

6.10 Ensure the following equipment NOT on backup power supply.

6.10.1 PY-15, 16, or 17 from TYBU (PER OP J-10:VII) (if PY-15, 16, or 17 is on backup, refer to OP AP-4)

6.10.2 Technical Support Center (OP O-13)

6.10.3 Spent Fuel Pool Pumps 1-1 OR 1-2 (OP O-13)

6.10.4 Control Room Vent Power Panel E-2 switch no. EPCE2 (OP O-13)

6.11 Ensure redundant equipment on other 480 VAC busses is running.

- Aux Bldg Switchgear Room Supply Fan S-44
- Aux Bldg Switchgear Room Exhaust Fan E-44

7. Loss of 480 VAC Bus G Supplemental Actions

NOTE: Reference DWG 437542 section 1C - PK17-22 Input 214.

7.1 IF DRPI was being supplied by backup power (52-1G-43), THEN perform the following:

7.1.1 Ensure Rod Control in manual and REFER TO TS 3.1.7.

7.1.2 Place DRPI on normal power PER OP A-3:I and OP O-13.

7.2 With Block Valve 8000B de-energized, within one hour, place PCV-455C hand switch to closed. REFER TO TS 3.4.11.

7.3 Assign personnel to perform necessary Nuclear Operator actions, steps 7.7 - 7.9 (battery charger transfer is a 2-hour action).

7.4 Ensure the following equipment NOT on backup power supply.

- Pressurizer Heater Group 1-2 (OP A-4A:I).

7.5 Ensure redundant equipment on the other 480 VAC busses is running.

- Primary Makeup Water Pump 1-1
- Boric Acid Transfer Pump 1-1 aligned to blender
- Makeup Water Transfer Pump 0-1
- Aux Building Supply Fan S-32
- Spent Fuel Pool Pump 1-2
- CFCUs 1-1, 1-2, and 1-4
- FHB Supply Fan S-2

7.6 Address the following issues as time permits.

7.6.1 RE-11 pump power off, reference OP H-2:I, Section 6.5 for actions.

7.6.2 IF both generator powered circuit breakers are open,
THEN ensure Main Turbine DC Bearing Oil Pump running.

7.6.3 Aux Feedwater supply LCVs 106, 107, 108, and 109 will require local operation.

7.6.4 Check status of vital instrument inverters.

NOTE: Diesel crankcase exhausters are desired, but not required, for starting and normal operation of the diesel generator. Reference AR A0503292.

7.6.5 DG 1-2 may be running without crankcase exhausters - consider shutting it down (Aux Motor Contactor Panel 'A').

7.6.6 ASW Pump may be running without exhaust fan cooling (E-101) if 4 kV bus still energized. Ensure correct continuous chlorination lineup.

7.6.7 Numerous valves fail as-is and will require local operation if necessary. See Fed From list for a complete listing (52-HG-10).

7.6.8 DG 1-1 Aux Motor Contactor Panel 'B' is lost.
- Consider installing air cross-tie hose PER OP J-6B:I.
- Monitor lube oil temperature (≥ 90°F, TS SR 3.8.1.2 Basis).

CAUTION: Diesel Generator margin assessment shall be completed at the earliest opportunity and if possible prior to transferring loads to their alternate vital bus to ensure the DG will not be overloaded during a loss of offsite power.

NOTE: With the exception of pressurizer heaters, all loads capable of being aligned to a vital bus as an alternate source can be aligned to the vital bus simultaneously provided that any ECCS pump powered from that bus is prevented from starting.

7.7 On any vital bus receiving additional load due to transfer of plant equipment to alternate source of power, perform diesel generator margin assessment PER OP O-13, "Transferring Equipment to/from Alternate Power Source." Reference SAPN 50350316.

7.8 IF DRPI was being supplied by backup power (52-1G-43),
THEN place DRPI on normal power PER OP A-3:I and OP O-13.

7.9 Transfer the following equipment to backup power as necessary.

- Battery 1-2 to Charger 1-21 PER OP J-9:II (TS 3.8.4, 3.8.5, 3.8.9, 3.8.10)
- Instrument AC PY-16 PER OP J-10:VII (reference OP AP-4)
- Control Room Vent Power Panel B-2 switch no. EPCB2 (OP O-13)

7.10 Ensure the following equipment NOT on backup power supply:

- Diesel Fuel Oil Transfer Pump 0-2 (OP O-13)

8. Loss of 480 VAC Bus H Supplemental Actions

NOTE:	Reference DWG 437543 section 1C - PK16-22 Input 288:

8.1 With Block Valve 8000C de-energized, within one hour place PCV-456 hand switch to closed. REFER TO TS 3.4.11.

8.2 Assign personnel to perform necessary Nuclear Operator actions, steps 8.6 - 8.9 (battery charger transfer is a 2-hour action).

8.3 Ensure the following equipment NOT on backup power supply.

- Pressurizer Heater Group 1-3 (OP A-4A:I)
- Battery 1-1 and 1-2 NOT on Battery Charger 1-21 (OP J-9:II)

8.4 Ensure redundant equipment on the other 480 VAC buses is running.

- Spent Fuel Pool Pump 1-1
- Aux Building Exhaust Fan E-1
- FHB Exhaust Fan E-4 or E-5
- CFCUs 1-1, 1-2, 1-3, or 1-5
- Makeup Water Transfer Pump 0-2
- FHB Supply Fan S-1
- Aux Bldg Supply Fan S-31

8.5 Address the following issues as time permits.

8.5.1 FWP 1-2 turning gear deenergized.

8.5.2 Aux feedwater supply LCVs 110 and 111 will require local operation.

8.5.3 Check status of vital instrument inverters.

NOTE:	Diesel crankcase exhausters are desired, but not required, for normal starting and operation of the Diesel Generator. Reference AR A0503292.

8.5.4 DG 1-1 may be running without crankcase exhausters - consider shutting it down (Aux Motor Contactor Panel 'A').

8.5.5 Numerous valves fail as is and will require local operation if necessary. See Fed From list for a complete listing (52-HH-10).

8.5.6 DG 1-3 Aux Motor Contactor Panel 'B' is lost.

- Consider installing air cross-tie hose PER OP J-6B:III.
- Monitor lube oil temperature (≥ 90°F, TS SR 3.8.1.2 Basis).

<u>CAUTION</u>:	Diesel Generator margin assessment shall be completed at the earliest opportunity and if possible prior to transferring loads to their alternate vital bus to ensure the DG will not be overloaded during a loss of offsite power.

<u>NOTE</u>:	With the exception of pressurizer heaters, all loads capable of being aligned to a vital bus as an alternate source can be aligned to the vital bus simultaneously provided that any ECCS pump powered from that bus is prevented from starting.

8.6 On any vital bus receiving additional loading due to transfer of plant equipment to an alternate source of power, initiate performance of diesel generator margin assessment per OP O-13, "Transferring Equipment to/from Alternate Power Source" (Reference SAPN 50350316).

8.7 Transfer the following equipment to backup power as necessary.

- Battery 1-3 to Battery Charger 1-31 PER OP J-9:II
- Instrument AC PY-15 PER OP J-10:VII (REFER TO OP AP-4)
- Diesel Fuel Oil Transfer Pump 0-1 PER OP O-13
- Communications Room power PER OP O-13
- Control Room Vent Power Panel D-2 switch no. EPCD2 PER OP O-13

8.8 Ensure the following equipment NOT on backup power supply.

- Control Room Vent Power Panel A-2 switch no. EPCA2 PER OP O-13

8.9 Ensure redundant equipment on the other 480 VAC buses is running.

- Aux Bldg Swgr Room Supply Fan S-43
- Aux Bldg Swgr Room Exhaust Fan E-43

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	1
APE 003 G2.4.23 Dropped Control Rod: Knowledge of the bases for prioritizing emergency operating procedures implementation	**Group #**	2
	K/A #	APE 003 G2.4.23
	Rating	4.4

Question 07

Unit 1 is at 75% power. The crew is raising power to 100% power.

As the operator withdraws the control rods, both a Control Bank C rod and a Control Bank D rod drop to the bottom of the core.

1. In accordance with OP AP-12C, Dropped Control Rod, the Shift Foreman should direct the crew to:
2. The reason for the action in OP AP-12C, is that:

A. 1. perform a plant shutdown in accordance with OP AP-25, Rapid Load Reduction or Shutdown.
   2. operation with more than one dropped rod is not an analyzed condition in the FSAR.

B. 1. perform a plant shutdown in accordance with OP AP-25, Rapid Load Reduction or Shutdown.
   2. FSAR analysis shows unacceptable power peaking factors may be produced.

C. 1. trip the reactor and go to EOP E-0, Reactor Trip or Safety Injection.
   2. operation with more than one dropped rod is not an analyzed condition in the FSAR.

D. 1. trip the reactor and go to EOP E-0, Reactor Trip or Safety Injection.
   2. FSAR analysis shows unacceptable power peaking factors may be produced.

Proposed Answer: C. 1. trip the reactor and go to EOP E-0, Reactor Trip or Safety Injection.
2. operation with more than one dropped rod is not an analyzed condition in the FSAR.

Explanation:

KA is met by knowing that in the case of more than one dropped rod, the EOPs action takes priority over the AOP.

A. Incorrect. First part is incorrect. Plausible because there are times OP AP-25 is entered, such as when shutting down for a tube leak or if power must be reduced because a dropped rod is not recovered within an hour. Second part is correct.
B. Incorrect. First part is incorrect, a reactor trip not a rapid shutdown is required. Second part is incorrect. Only one rod drop is analyzed in the FSAR. Plausible because according to the bases for LCO 3.1.4, this is a concern for misaligned rods if the LCO is not met.
C. Correct. OP AP-12C directs a plant trip if there is more than one dropped rod. This is because operation with only 1 dropped rod is an analyzed accident in the FSAR. Two dropped rods is not analyzed.
D. Incorrect. First part is correct. OP AP-12C directs a reactor trip. Second part incorrect, the FSAR only analyzes a single dropped rod at power, therefore, its an unanalyzed

condition. Plausible because this is a stated potential problem with misaligned rods is the bases for LCO 3.1.4

Technical References: OP L-2, OP AP-12C, LPA12, FSAR Chapter 15.2, LCO B3.1.4

References to be provided to applicants during exam: None

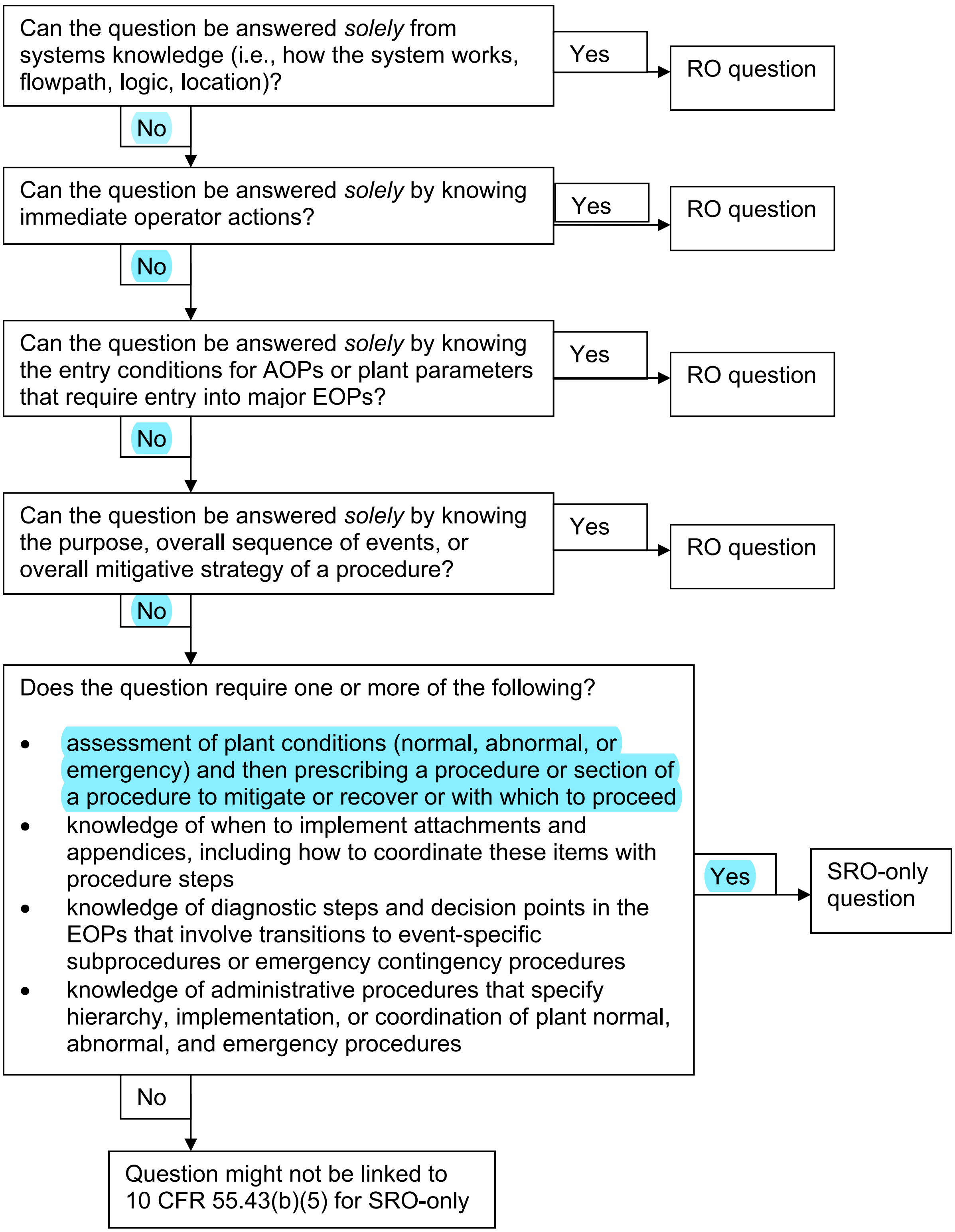
Learning Objective: Discuss abnormal conditions associated with the Rod Control System. (9903)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	N
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 2.5

Question 7

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



OP AP-12C, Major Actions and Procedure Review

Major Actions
Obj 4, 5, 10

Major actions of OP AP-12C are:

- *If more than one rod has dropped, trip the Reactor and GO TO EOP E-0.*
- Stabilize plant: stop load changes, stop any boration/dilution in progress, match T_{AVG} - T_{REF} within 1.5°F.
  Check AFD and perform STPs R-25 & R-19.
- Determine failure and correct cause of dropped rod.
- Reduce power to < 90%, or < 50% if > 1 hour since dropped rod event.
- Withdraw dropped rod and restore plant to normal.

OP AP-12C Review
Obj 5, 6, 8, 9, 10

Review OP AP-12C with the students and emphasize the following items.

- Requirements for tripping reactor and shutting down unit.
- Tech Spec requirements.
- Other discussion items in the list below:

Item	Discussion
Trip the reactor if > 1 rod has dropped. (*Obj 5, 10*)	• Operation with > 1 dropped rod is NOT analyzed.
Stabilize plant with rods in manual and use turbine load to match T_{AVE} and T_{REF} or adjust boron, if necessary.	Excessive mismatch may result in localized power peaking and xenon distribution effects. (*Obj 8*) It will be easier to determine SDM if the RCS has not been diluted.
Fully inset control banks if reactor is subcritical.	Startup mode is not analyzed with one dropped rod (rod misaligned).
Check Axial Flux Difference within T.S. limits.	• TS 3.2.3 • reduce power to < 50% w/in 30 min
Calculate QPTR per STP R-25. • Verify less than 1.02	• If > 1.02 then refer to TS 3.2.4.A
Ensure SDM within limits per STP R-19	• If no, refer to TS 3.1.1 / 3.1.4 • If required, emergency borate to restore SDM within 15 minutes.
Do NOT attempt to move rods or reset alarm if Urgent Failure is in.	• Attempting to move rods may cause more rods to drop and/or worsen conditions. • Resetting alarm clears local alarm indicators and may make it harder for TM to find fault(s). This may remove power to the CRDMs if powered by the moveable coils.

Continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. ONLY One Control Rod Dropped	Trip the Reactor and GO TO EOP E-0, REACTOR TRIP OR SAFETY INJECTION
2. PLACE Rod Control in MANUAL	
3. STOP Any Load Change In Progress AND Allow Conditions To Stabilize:	
a. Reactor is critical.	a. Place the reactor in Mode 3 by fully inserting the control rods. GO TO OP L-5, PLANT COOLDOWN FROM MINIMUM LOAD TO COLD SHUTDOWN

NOTE: The initial determination of SDM (STP R-19) is simplified if dilution is not used to raise temperature.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. STOP Any RCS Boration or Dilution In Progress	
5. ADJUST Turbine Load To Match T_{AVG} AND T_{REF} to Within ± 1.5°F	
6. CHECK Axial Flux Difference Within Tech Spec Limits	Refer to Tech Spec 3.2.3.
7. Calculate QPTR per STP R-25:	
a. Ensure LESS than 1.02	a. Refer to Tech Spec 3.2.4.A.
8. ENSURE SDM within COLR limits:	
a. Perform STP R-19 within 1 hour	a. Refer to Tech Spec 3.1.1 and 3.1.4
9. ENSURE Rod Control System Had No Urgent Failure:	
a. Ensure Rod Cont Urgent Failure (PK03-17) - OFF	a. If PK03-17 is ON, THEN 1. Do not attempt to move rods or reset the Urgent Failure. 2. Contact maintenance for trouble shooting. 3. Refer to AR PK03-17, ROD CONT URGENT FAILURE.

Power may be reestablished either by reactivity feedback or control bank withdrawal. Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control system interaction is monotonic, thus removing power overshoot as a concern and establishing the automatic rod control mode of operation as the limiting case.

For a dropped RCCA event in the automatic rod control mode, the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power. Figures 15.2.3-1 and 15.2.3-2 show a typical transient response to a dropped RCCA(s) in automatic control. Uncertainties in the initial conditions are included in the DNB evaluation as described in Reference 10. In all cases, the minimum DNBR remains above the safety analysis limit value.

Following plant stabilization, the operator may manually retrieve the RCCA(s) by following approved operating procedures.

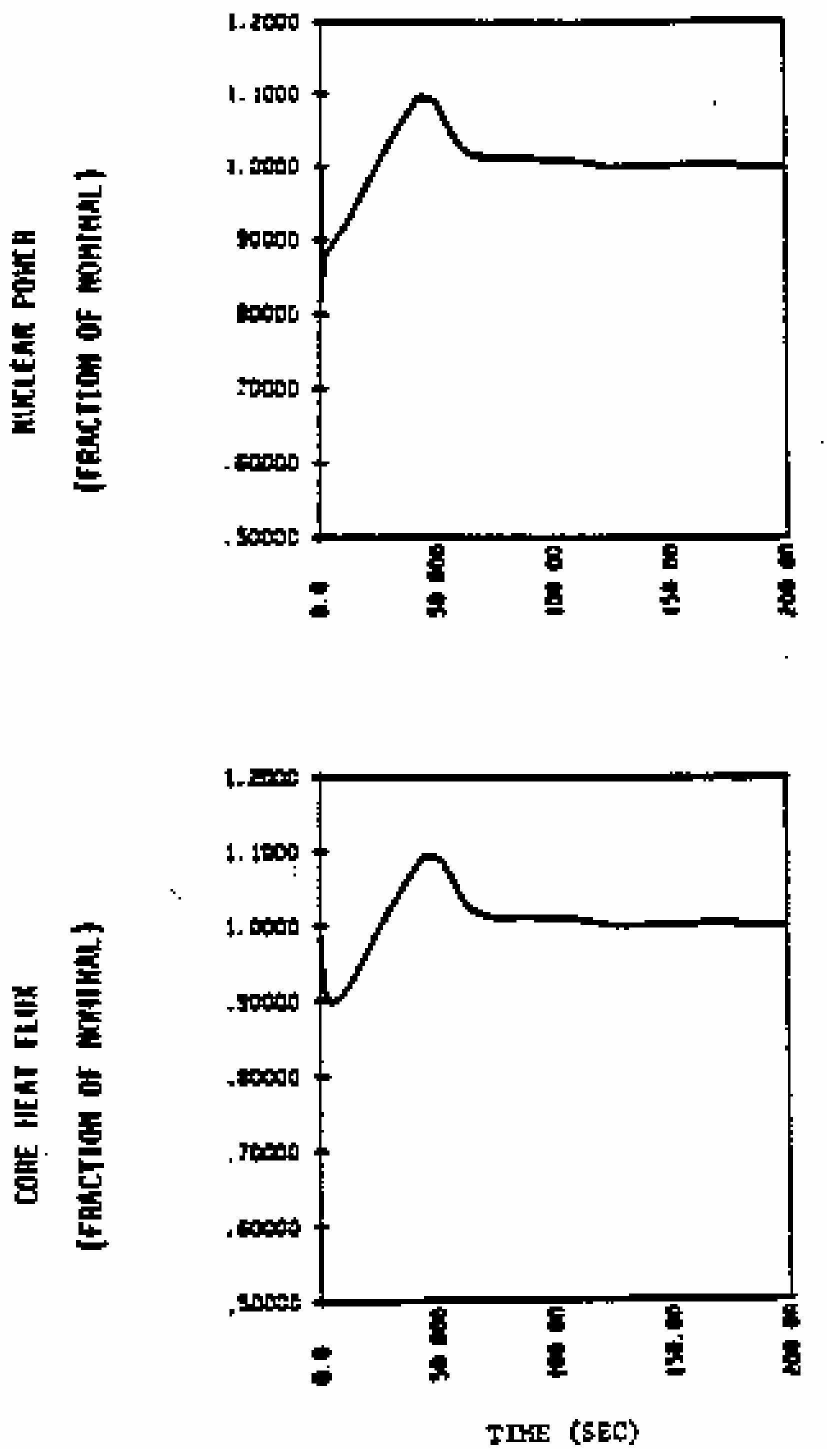
(2) Dropped RCCA Bank

A dropped RCCA bank typically results in a reactivity insertion of greater than 500 pcm. The core is not adversely affected during the insertion period since power is decreasing rapidly. The dropped RCCA bank transient will proceed as described in the previous section for one or more dropped RCCA(s), except the return to power will be less due to the greater worth of the entire bank. The power transient for a dropped RCCA bank is symmetric. Following plant stabilization, normal procedures are followed.

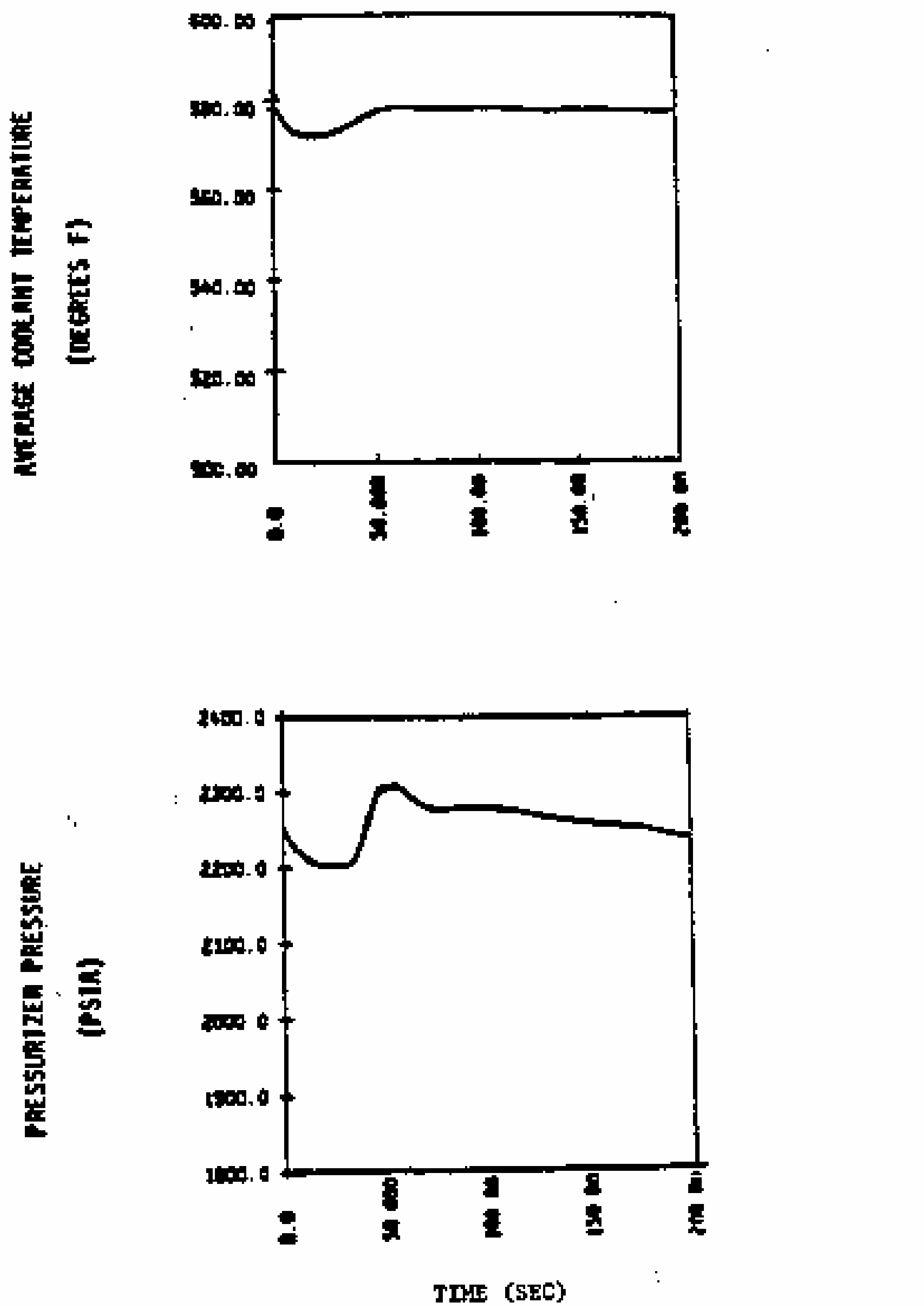
(3) Statically Misaligned RCCA

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCCA is fully inserted, or where Bank D is fully inserted with one RCCA fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the postulated conditions are approached. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the limit value.

The insertion limits in the Technical Specifications may vary from time to time depending on a number of limiting criteria. The full power insertion limits on control bank D must be chosen to be above that position which meets the minimum DNBR and peaking factor limits. The full power



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.3-1 **TRANSIENT RESPONSE TO DROPPED ROD CLUSTER CONTROL ASSEMBLY**



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.2.3-2
TRANSIENT RESPONSE TO DROPPED ROD CLUSTER CONTROL ASSEMBLY

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops — MODES 1 and 2

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the RCS include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;

b. Improving the neutron economy by acting as a reflector;

c. Carrying the soluble neutron poison, boric acid;

d. Providing a second barrier against fission product release to the environment; and

e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

The reactor coolant is circulated through four loops connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow and temperature instrumentation for both control and protection. The reactor vessel contains the clad fuel. The SGs provide the heat sink to the isolated secondary coolant. The RCPs circulate the coolant through the reactor core and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. This forced circulation of the reactor coolant ensures mixing of the coolant for proper boration and chemistry control.

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the design bases accident initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service.

All of the accident/safety analyses performed at RTP assume that all four RCS loops are in operation as an initial condition. Some accident/safety analyses have been performed at zero power conditions assuming only two RCS loops are in operation to conservatively bound lower modes of operation. The uncontrolled Rod Control Cluster Assembly (RCCA) Bank withdrawal from subcritical event is included in this category. While all accident/safety analyses performed at full rated power assume that all RCS loops are in operation, selected events examine the effects resulting from a loss of RCP operation. These include the complete and partial loss of forced

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	RCS flow, RCP rotor seizure, and RCP shaft break events. For each of these events, it is demonstrated that all the applicable safety criteria are satisfied. For the remaining accident/safety analyses, operation of all four RCS loops during the transient up to the time of reactor trip is assured thereby ensuring that all the applicable acceptance criteria are satisfied. Those transients analyzed beyond the time of reactor trip were examined assuming that a loss of offsite power occurs which results in the RCPs coasting down. The plant is designed to operate with all RCS loops in operation to maintain DNBR above the Safety Limit value during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant. RCS Loops - MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36 (c)(2)(ii).
LCO	The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power. An OPERABLE RCS loop consists of one OPERABLE RCP for heat transport and the associated OPERABLE SG, with a water level within the limits specified in SR 3.4.5.2, except for operational transients. A RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.
APPLICABILITY	In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage. The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

(continued)

Examination Outline Cross-Reference

APE 024 AA2.04 Ability to determine and/or interpret the following as they apply to Emergency Boration: Availability of the BWST

Level	SRO
Group #	1
K/A #	APE 024 AA2.04
Rating	3.7

Question 08

GIVEN:

- Unit 1 tripped from 100% power
- RCS Tave is 547°F
- RCS boron concentration is 300 ppm
- 2 Control Rods are stuck out
- Boric Acid Storage Tank (BAST) #1 is in service and level is 96%
- Boric Acid Storage Tank (BAST) #2 level is 96%
- RWST level is 96%

The crew initiates emergency boration through the preferred flow path in accordance with OP AP-6, Emergency Boration.

When the required boration is complete, what action should be taken?

NOTE: assume a 1 ppm change requires 9 gallons of 4% boric acid

A. Within one hour, restore the RWST to OPERABLE status.

B. Within 72 hours, restore the Boric Acid Storage system to OPERABLE status.

C. Makeup to the BAST but no Technical Specification or ECG ACTION is required.

D. Makeup to the RWST but no Technical Specification or ECG ACTION is required.

Proposed Answer: B. Within 72 hours, restore the Boric Acid Storage system to OPERABLE status.

Explanation:
NOTE: thumb rule was recently removed from OP AP-6, given in question for consistency of calculation.
The SRO must determine the following:
Amount of ppm change – AP-6 states: Borate 400 ppm per stuck rod OR borate to 2000 - 2400 ppm boron concentration, whichever is less. (for current plant condition, 300 ppm +800 ppm = 1100 ppm final concentration).
The "normal source" is in OP AP-6 – makeup using the CVCS makeup controller. Thumb rule (GFES) for a ppm change - 9 gallons/ppm for BAST. (This correlates to approximately 27 gallons/ppm if using the RWST - about a 3/1 ratio due to differences in boron concentration)
Amount of inventory required. #stuck rods x required ppm change x thumb rule = 2 x 400 ppm x
9 gals/ppm = 7200 gallons (21600 is assuming RWST) Change in BAST level = (95% level) 14964 gallons – 7200 = 7764 gallons remaining which is less than the requirement in ECG 8.9.

A. Incorrect. If its assumed the RWST are the alternate source and the math correctly calculated, this would be the answer. 463354 – 21600 = 441934 gallons.

B. Correct. The required level of 14042 gallons will not be met and the LCO action must be taken to restore the BAST to OPERABLE status within 72 hours
C. Incorrect. The combined levels of the BAST is 14964 gallons. If the conversion of ppm to gallons is not done and 800 is used, then the assumed level would be 14114 gallons. Still above the requirements of ECG 8.9
D. Incorrect. If the thumb rule for the BAST is used, then 7200 gallons would be the calculated change, which is still above the required 455300 gallons of LCO 3.5.4 (463534 – 7200 = 456334 gallons)

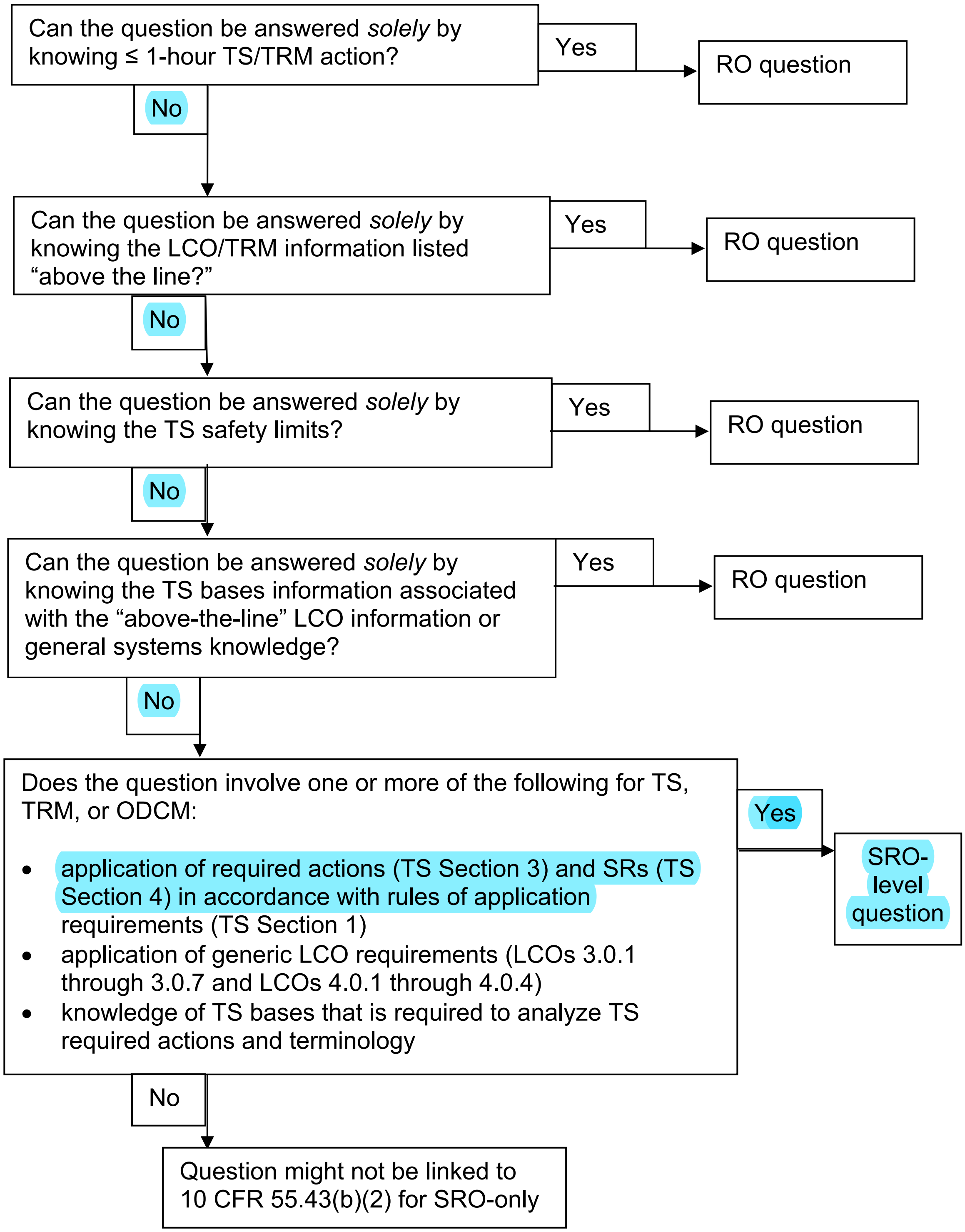
Technical References: ECG 8.9, Tech Spec 3.5.4, STP C-20, OP AP-6
References to be provided to applicants during exam: ECG 8.9 (LCO only), Tech Spec 3.5.4,
BAST and RWST level tables from C-20
Learning Objective: 66041 - Apply the requirements of System 8 ECGs

Question Source:	Bank #82 L141 NRC	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 04/2016	Y
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.2	

Difficulty: 3

Question 8

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



c. *Facility Licensee Procedures Required to Obtain Authority for Design and Operating Changes in the Facility [10 CFR 55.43(b)(3)]*

Some examples of SRO-only examination items for this topic include the following:

Examination Outline Cross-Reference	Level	SRO
APE024 AA2.04- Ability to determine and interpret the following as they apply to the Emergency Boration: Availability of BWST (BAST - DCPP)	**Tier #**	1
	Group #	2
	K/A #	APE 024 AA2.04
	Rating	4.2

Question 82 PARENT QUESTION

GIVEN:
- Unit 1 tripped from 100% power
- RCS Tave is 547°F
- RCS boron concentration is 300 ppm
- 2 Control Rods are stuck out
- Boric Acid Storage Tank (BAST) #1 is in service and level is 96%
- Boric Acid Storage Tank (BAST) #2 level is 96%
- RWST level is 96%

The crew initiates emergency boration through the preferred flow path in accordance with OP AP-6, Emergency Boration.

When the required boration is complete, what ECG or Technical Specification action, if any, should be taken?

NOTE: assume a 1 ppm change requires 9 gallons of 4% boric acid

A. Makeup to the BAST but no Technical Specification or ECG ACTION is required.

B. Makeup to the RWST but no Technical Specification or ECG ACTION is required.

C. Within one hour, restore the RWST to OPERABLE status.

D. Within 72 hours, restore the Boric Acid Storage system to OPERABLE status.

Proposed Answer: D. Within 72 hours, restore the Boric Acid Storage system to OPERABLE status.

Explanation:

The SRO must determine the following:
Amount of ppm change – AP-6 states: Borate 400 ppm per stuck rod OR borate to 2000 - 2400 ppm boron concentration, whichever is less. (for current plant condition, 300 ppm + 800 ppm = 1100 final concentration).
What the "normal source" is in OP AP-6 – makeup using the CVCS makeup controller.
Thumb rule (GFES) for a ppm change - 9 gallons/ppm for BAST. (This correlates to approximately 27 gallons/ppm if using the RWST - about a 3/1 ratio due to differences in boron concentration)
Amount of inventory required. #stuck rods x required ppm change x thumb rule = 2 x 400 ppm x 9 gals/ppm = 7200 gallons (21600 is assuming RWST)
Change in BAST level = (95% level) 14964 gallons – 7200 = 7764 gallons remaining which is less than the requirement in ECG 8.9.

A. Incorrect. The combined levels of the BAST is 14964 gallons. If the conversion of ppm to gallons is not done and 800 is used, then the assumed level would be 14114 gallons. Still above the requirements of ECG 8.9
B. Incorrect. If the thumb rule for the BAST is used, then 7200 gallons would be the calculated change, which is still above the required 455300 gallons of LCO 3.5.4 (463534 – 7200 = 456334 gallons)
C. Incorrect. If its assumed the RWST are the alternate source and the math correctly calculated, this would be the answer. 463354 – 21600 = 441934 gallons
D. Correct. The required level of 14042 gallons will not be met and the LCO action must be taken to restore the BAST to OPERABLE status within 72 hours.

Technical References: ECG 8.9, Tech Spec 3.5.4, STP C-20, OP AP-6.
References to be provided to applicants during exam: ECG 8.9 (LCO only), Tech Spec 3.5.4, BAST and RWST level tables from C-20
Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.2	

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RWST boron concentration not within limits. OR RWST borated water temperature not within limits.	A.1 Restore RWST to OPERABLE status.	8 hours
B. RWST inoperable for reasons other than Condition A.	B.1 Restore RWST to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 ------------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------- Be in MODE 4.	6 hours 12 hours

8.0 CHEMICAL AND VOLUME CONTROL SYSTEM

8.9 Reactivity Control Systems - Borated Water Source - Operating

ECG 8.9 The Boric Acid Storage System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boric Acid Storage System inoperable.	A.1 Restore Boric Acid Storage System to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in at least MODE 3.	6 hours
	AND	
	B.2 Verify SDM is within the limits provided in the COLR*.	6 hours
	AND	
	B.3 Restore Boric Acid Storage System to OPERABLE status.	7 days
C. Required Action and associated Completion Time of Condition B.3 not met.	C.1 Be in MODE 5	30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 8.9.1	Verify Boric Acid Storage System boron concentration is ≥ 7,000 ppm and ≤ 7,700 ppm	7 days
SR 8.9.2	Verify Boric Acid Storage System usable borated water volume is ≥ 14,042 gallons.	7 days
SR 8.9.3	Verify Boric Acid Storage System solution temperature is ≥ 65°F.	7 days

* Core Operating Limits Report (COLR)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.4.1	------------------------------NOTE---------------------------- Only required to be performed when ambient air temperature is < 35°F. --- Verify RWST borated water temperature is $\geq$ 35°F.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.2	Verify RWST borated water volume is $\geq$ 455,300 gallons.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.3	Verify RWST boron concentration is $\geq$ 2300 ppm and $\leq$ 2500 ppm.	In accordance with the Surveillance Frequency Control Program

Usable volume, gallons = ((Indicated level in % minus 3%) x 65.37)) + 1566 - 163

% LEVEL	VOLUME (GAL)
0 - 3	≤ 1403
4	1468
5	1534
6	1599
7	1664
8	1730
9	1795
10	1861
11	1926
12	1991
13	2057
14	2122
15	2187
16	2253
17	2318
18	2384
19	2449
20	2514
21	2580
22	2645
23	2710
24	2776
25	2841
26	2907
27	2972
28	3037
29	3103
30	3168
31	3233
32	3299
33	3364
34	3429
35	3495
36	3560
37	3626
38	3691
39	3756
40	3822
41	3887
42	3952
43	4018
44	4083
45	4149
46	4214
47	4279
48	4345
49	4410
50	4475
51	4541

% LEVEL	VOLUME (GAL)
52	4606
53	4672
54	4737
55	4802
56	4868
57	4933
58	4998
59	5064
60	5129
61	5194
62	5260
63	5325
64	5391
65	5456
66	5521
67	5587
68	5652
69	5717
70	5783
71	5848
72	5914
73	5979
74	6044
75	6110
76	6175
77	6240
78	6306
79	6371
80	6436
81	6502
82	6567
83	6633
84	6698
85	6763
86	6829
87	6894
88	6959
89	7025
90	7090
91	7156
92	7221
93	7286
94	7352
95	7417
96	7482
97	7548
98	7613
99	7679
100	7744

Usable volumes in the above table are based on a level indicator uncertainty of 3%.

Emergency Boration

03/26/21
Effective Date

QUALITY RELATED

1. SCOPE

1.1 This procedure covers situations which require emergency boration and the methods for accomplishing this operation. Various options of emergency boration are discussed in this procedure.[T36122]

1.2 The boration methods in order of preference are as follows. The last method, CVCS-8471, is too involved and takes so much time that it is only used as the last option. Emergency boration flowrate is not specifically defined. TS 3.1.1 Basis states the Operator should borate with the best source available for plant conditions.

- VCT Makeup
- CVCS-8104, Emergency Boration Valve
- RWST
- CVCS-8471, Manual Emergency Boration Valve

2. SYMPTOMS OR ENTRY CONDITIONS

2.1 Control or shutdown rods inserted below the low-low insertion limit when critical. Commence boration within 1 hour and restore rods to above the Lo-Lo Insertion Limit within 2 hours, (TS 3.1.5 & 3.1.6) ROD LO LO INSERTION LIMIT (PK03-14).

2.2 Failure of 2 or more control rods to fully insert following a reactor trip as indicated by rod position indication and rod bottom lights or other indications of a lack of reactivity control. Commence boration when directed by Emergency Procedures.

2.3 Uncontrolled RCS cooldown following a reactor trip with no ESF actuation (EOP E-0.1).

2.4 Uncontrolled or unexplained reactivity rise as indicated by:

- Unexplained control rod insertion
- Rising T_{AVG} or nuclear power with no rising load demand
- Unexpected rising count rate when shutdown

2.5 When boration is required and normal boration through the VCT makeup system is not possible. Borate as required per SFM direction.

2.6 Shutdown margin less than acceptable minimum limits per TS 3.1.1 and 3.9.1. Commence boration within 15 minutes to restore rods to restore shutdown margin.

APPENDIX A

Boration Requirements

Borate prescribed amount according to entry symptom:

Symptom	Boration Requirement
1. Rods below RIL while critical.	1. Commence boration within 1 hour and restore rods to above the RIL within 2 hours.
2. Two or more stuck rods following reactor trip.	2. Borate 400 ppm per stuck rod OR borate to 2000 - 2400 ppm boron concentration, whichever is less.
3. Uncontrolled cooldown following reactor trip without ESF Actuation.	3. Borate until adequate shutdown margin is attained.
4. Uncontrolled or unexplained reactivity rise.	4. Borate until control regained. REFER TO EOP FR-S.1 APPENDIX D to isolate dilution flowpaths, if required.
5. When normal boration methods unavailable.	5. Borate as required to maintain proper boron concentration.
6. SDM less than acceptable.	6. Within 15 minutes, commence and maintain boration until adequate SDM is attained.

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	2
APE 051 AA2.02 Ability to determine and/or interpret the following as they apply to Loss of Condenser Vacuum: Conditions requiring reactor and/or turbine trip	**Group #**	1
	K/A #	APE 051 AA2.02
	Rating	4.2

Question 09

GIVEN:
- The crew is raising turbine load at 3 MWe/minute in accordance with OP L-4, Normal Operation at Power
- Turbine Load is 400 Mwe

The operator reports the following:
- Condenser Pressure Recorder PR-11A and B both show condenser pressure is slowly rising.
- Condenser Pressure PI-44 reads 8.3" Hg Abs and rising slowly
- Condenser differential pressure is 6.9 psid and rising slowly on all quadrants
- PK10-11 CONDENSER PRESS/LEVEL is in alarm

Which of the following actions should be taken by the Shift Foreman?

Go to OP AP-7, Degraded Condenser ______1.______ and in accordance with the procedure, direct the crew to ______2.______.

A. 1. Section A, Loss of Condenser Vacuum
   2. trip the turbine and go to OP AP-29, Main Turbine Malfunction

B. 1. Section A, Loss of Condenser Vacuum
   2, trip the reactor and go to EOP E-0, Reactor Trip or Safety Injection

C. 1. Section B, Condenser Fouling
   2. reduce load to remove a Circulating Water pump from operation

D. 1. Section B, Condenser Fouling
   2. trip the reactor and go to EOP E-0, Reactor Trip or Safety Injection

Proposed Answer: A. 1. Section A, Loss of Condenser Vacuum
2. trip the turbine and go to OP AP-29, Main Turbine Malfunction

Explanation: SRO must assess plant conditions to determine which section of OP AP-7 entry is required due to degrading vacuum and dp. Entry into both sections is possible. DP values are higher than the normal values, which are typically 4 to 5 psid, however, operationally, entering section B would not address the more important issue of higher than allowable vacuum pressure and a trip is necessary, nor does Section B provide a path back to the section A to address the vacuum issue. Then understand that condenser pressure is too high and the turbine (not the reactor) must be tripped – based on power level, then determine if a reactor trip and E-0 entry or turbine trip and AP-29 entry required

A. Correct. Condenser pressure is above the allowable limit (7.2 inches Hg abs) until power is above 50% (600 Mwe). Power is less than 50%, P-9 is not enabled. Action is to trip the turbine and go to OP AP-29.
B. Incorrect. First part is correct. Power is less than 50%, action is to trip the turbine. Plausible because this would be correct if power was above 50%.
C. Incorrect. Section A must be performed. Plausible because dp is elevated, it is not high enough to warrant attention. Second part is plausible as this is the action if a CWP must be removed from service due to high quadrant dp.
D. Incorrect. Both parts are incorrect. The dp values, while elevated do not warrant action. Section A applies. Second part is plausible, this is the action if there is one CWP operating above 25% and it must be tripped.

Technical References: OP AP-7 section A and B
References to be provided to applicants during exam: None
Learning Objective: Given initial conditions and assumptions, determine if a turbine trip is required. (7968)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #08 L091C	X
	New	
	Past NRC Exam DCPP 03/2012	Y
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 3

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)

Can the question be answered *solely* from systems knowledge (i.e., how the system works, flowpath, logic, location)?

Yes → RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No

Can the question be answered *solely* by knowing the entry conditions for AOPs or plant parameters that require entry into major EOPs?

Yes → RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No

Does the question require one or more of the following?

- assessment of plant conditions (normal, abnormal, or emergency) and then prescribing a procedure or section of a procedure to mitigate or recover or with which to proceed
- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific subprocedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Examination Outline Cross-Reference	Level	SRO
APE 051 AA2.01 Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum: Cause for low vacuum condition	**Tier #**	1
	Group #	2
	K/A #	APE051 AA2.01
	Rating	2.7

Question SRO 08 (83) PARENT QUESTION

GIVEN:
- A turbine load increase is in progress with all systems operating normally in automatic control
- Turbine Load is 400 MWe

The operator reports the following:
- Condenser Pressure Recorder PR-11A and B both show condenser pressure is slowly rising.
- Condenser Pressure PI-44 reads 4.0"Hg Abs
- Condenser differential pressure is 7 psid and stable on all quadrants
- PK10-11 CONDENSER PRESS/LEVEL is in alarm

Which of the following actions should be taken by the Shift Foreman?

A. Direct the operator to trip the turbine and go to AP-29, Main Turbine Malfunction, due to high condenser pressure.

B. Direct the operator to trip the turbine and go to AP-29, Main Turbine Malfunction, due to high quadrant DP.

C. Go to OP AP-7, Degraded Condenser, section A, Loss of Condenser Vacuum, and reduce load as necessary to restore condenser pressure to within operating limits.

D. Go to OP AP-7, Degraded Condenser, section B, Condenser Fouling, and reduce load to remove a Circulating Water pump from operation to lower condenser differential pressure.

Proposed Answer: C. Go to OP AP-7, Degraded Condenser, section A, Loss of Condenser Vacuum, and reduce load as necessary to restore condenser pressure to within operating limits.

Explanation:
A. Incorrect. The nominal setpoint for tripping on low vacuum is 7.2 inches Hg.
B. Incorrect. High quadrant DP trip setpoint is any DP greater than 10 psid and increasing rapidly or, all halves greater than 10 or any greater than 13 psid.
C. Correct. The problem is condenser vacuum, the action is to reduce load in an effort to stabilize vacuum.
D. Incorrect. DP is stable

Technical References: AR PK10-11, OP AP-7, section A and B
References to be provided to applicants during exam: None
Learning Objective: 3477G Given an abnormal condition, summarize the major actions of OP AP-7 to mitigate an event in progress.

Question Source:	Bank # 83 DCPP NRC Exam 1/2010	X
(note changes; attach parent)	Modified Bank #	

	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.	

SECTION A: LOSS OF CONDENSER VACUUM (Continued)

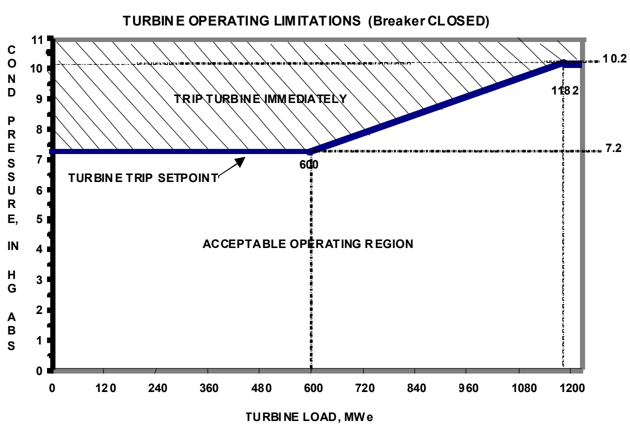
3. INSTRUCTIONS

ACTION/EXPECTED RESPONSE | **RESPONSE NOT OBTAINED**

CAUTION 1: If both CWPs are lost, Attachment 1 and Attachment 4 should be implemented after EOP E-0 is performed.

CAUTION 2: If it appears Operator actions to stabilize the plant will not be successful, manually trip the reactor and GO TO EOP E-0.[T35691]

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **CHECK Condenser** • Condenser Pressure LESS THAN maximum allowed PER Figure 1 (also in Attachment 2)	a. IF GREATER THAN P-9, THEN trip the Reactor AND GO TO EOP E-0. b. IF reactor power LESS THAN P-9, THEN perform the following: 1) INITIATE turbine trip. 2) IMPLEMENT remainder of this procedure. 3) GO TO OP AP-29, MAIN TURBINE MALFUNCTION.



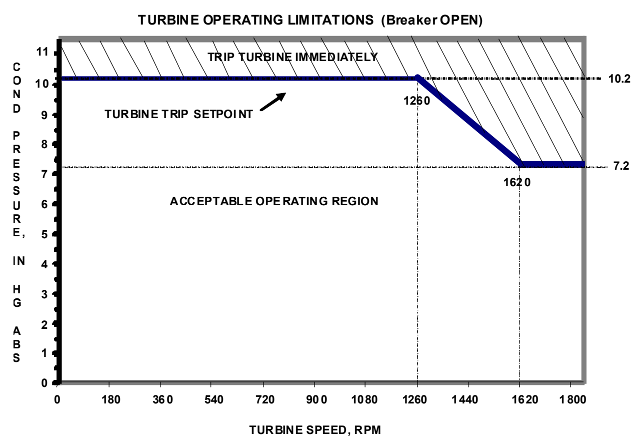


Figure 1: Turbine Operating Limitations

SECTION A: LOSS OF CONDENSER VACUUM (Continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. STABILIZE Condenser Pressure:	
a. Reduce unit load as necessary to maintain Condenser pressure within the limitations of Attachment 2[T35691]	a. IF operating limitations are exceeded, THEN perform the following: 1) IF reactor power is GREATER THAN P-9, THEN trip the Reactor AND GO TO EOP E-0. 2) IF reactor power LESS THAN P-9, THEN perform the following: a) INITIATE a turbine trip. b) IMPLEMENT remainder of this procedure. c) GO TO OP AP-29, MAIN TURBINE MALFUNCTION.
3. ANNOUNCE loss of condenser vacuum on PA	
4. CHECK Condenser pressure - LESS THAN 10.2 inches Hg (absolute)	a. IF GREATER THAN 15% power, THEN perform the following: 1) Trip the reactor 2) Trip BOTH MFW pumps: • MFW Pump 1-1 (2-1) • MFW Pump 1-2 (2-2) 3) GO TO EOP E-0. b. IF LESS THAN 15% power, THEN perform the following: 1) Trip BOTH MFW pumps: • MFW Pump 1-1 (2-1) • MFW Pump 1-2 (2-2) 2) IMPLEMENT remainder of this procedure. 3) GO TO OP AP-15, Section B.

SECTION B: CONDENSER FOULING

2. SYMPTOMS

2.1 Rising Condenser pressure as indicated on:

- PI-44, digital readout on VB3 (0-50 inches Hg absolute), which is fed from the MSS from PIT-25/26/27 and also displayed on the Turbine DEH Tricon.
- PR-11A, B Condenser pressure recorder on VB3 (0-10 inches Hg absolute), are fed from PT-175 and PT-176 located on the 104' and are susceptible to false indications due to moisture in the sensing lines.
- Main Turbine Control screen "Main Turbine Overview"

2.2 Rising Condenser Differential Pressure (PPC GRPDIS PK13-04 and/or PPC trend for DP Rate of Change)

2.3 Possible lowering generator output or rise in reactor power

2.4 CWP amps oscillating

SECTION B: CONDENSER FOULING (Continued)

3. INSTRUCTIONS

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**

CAUTION 1: If both CWPs are lost, Attachment 1 and Attachment 4 should be implemented after EOP E-0 is performed.

CAUTION 2: The backflow from shutdown of an affected CWP could cause a large rise in loading on the screens of the CWPs still in service. The screens for the unaffected pumps should be in high speed prior to the CWP shutdown.

NOTE: Action should be taken to secure CWPs when hard limits below are reached; however, preemptive actions may be taken at any time per Ops Management discretion.

1. **CHECK Condenser quadrant differential pressures - ABNORMAL:** (PPC GRPDIS PK13-04)
   - Any quadrant ΔP GREATER THAN 10 psid AND RISING rapidly
   - All Condenser ΔPs GREATER THAN 10 psid with known external debris event in progress
   - All condenser ΔPs GREATER THAN 12 psid
   - Any ΔP at 14 psid

   Response Not Obtained: RETURN to procedure and step in effect.

2. **Dispatch Operator to Intake**
   - Monitor screen performance
   - Monitor CWP for cavitation

3. **MONITOR CWP(s) - No indication of cavitation**
   - Amps NORMAL - approximately 560 amps and LESS THAN 20 amp swings
   - No excessive pump vibrations/noise

   Response Not Obtained: Expedite ramp to remove CWP from service.

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	2
E14 G2.2.44 Loss of Containment Integrity: 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	**Group #**	1
	K/A #	E14 G2.2.44
	Rating	4.4

Question 10

GIVEN:

- A design basis LOCA occurs from 100% power
- The crew enters EOP E-0, Reactor Trip or Safety Injection
- When the operator begins to perform Appendix E, ESF Auto Actions, Secondary and Auxiliaries Status, the following conditions exist:
    - Only Bus F is energized
    - Containment pressure has risen to 25 psig
    - RCS cold leg temperatures are 300°F

At step 11 of Appendix E, REPORT to SFM, the operator states all 4 kV Vital Bus F equipment has been started or verified running and steam generator narrow range levels are as follows:

- 1-1 Steam Generator – 19%
- 1-2 Steam Generator – 19%
- 1-3 Steam Generator – 26%
- 1-4 Steam Generator – 22%

When the Shift Foreman exits EOP E-0, the procedure to be entered should be:

A. EOP E-1, Loss of Reactor or Secondary Coolant

B. EOP FR-H.1, Response to Loss of Secondary Heat Sink

C. EOP FR-P.1, Response to Imminent Pressurized Thermal Shock Condition

D. EOP FR-Z.1, Response to High Containment Pressure

Proposed Answer: D. EOP FR-Z.1, Response to High Containment Pressure

Explanation:
The SRO must know what actions are taken by the operator in Appendix E and power supplies to vital 4 kV loads, then evaluate plant conditions to make the proper procedure selection.
Bus F will energize pumps
ASW pump 1-1
AFW pump 1-3
CCP 1-1
CCW pump 1-1
SI pump 1-1

A. Incorrect. Per F-0, there will be a Magenta path on Containment Integrity status tree that requires addressing prior to performing EOP E-1. Plausible because there is a LOCA occurring.

B. Incorrect. There will be an AFW pump started in Appendix E by the operator; AFW pump 1-3 is powered from bus F. Also, with adverse containment, a steam generator level of 25% is required – but only in one steam generator. If its thought that there must be 3 steam generators above the required level (as required to shutdown the TDAFW pump, and therefore, a RED path will exist for Heat Sink CSF, then this would be the answer. 3 steam generators low (18% WR or 26% WR adverse) is the criteria for feed and bleed in H.1.
C. Incorrect. RCS temperatures are high enough that RCS integrity status tree is YELLOW.
D. Correct. Both Containment spray pumps will not be able to be started (powered from Bus G and H). With Containment pressure above 22 psig, the Containment Integrity Status tree is MAGENTA and entry into EOP FR-Z.1 is warranted.

Technical References: OIM J-1-1, EOP E-0 Appendix E, F-0, EOP FR-H.1 Foldout page

References to be provided to applicants during exam: None

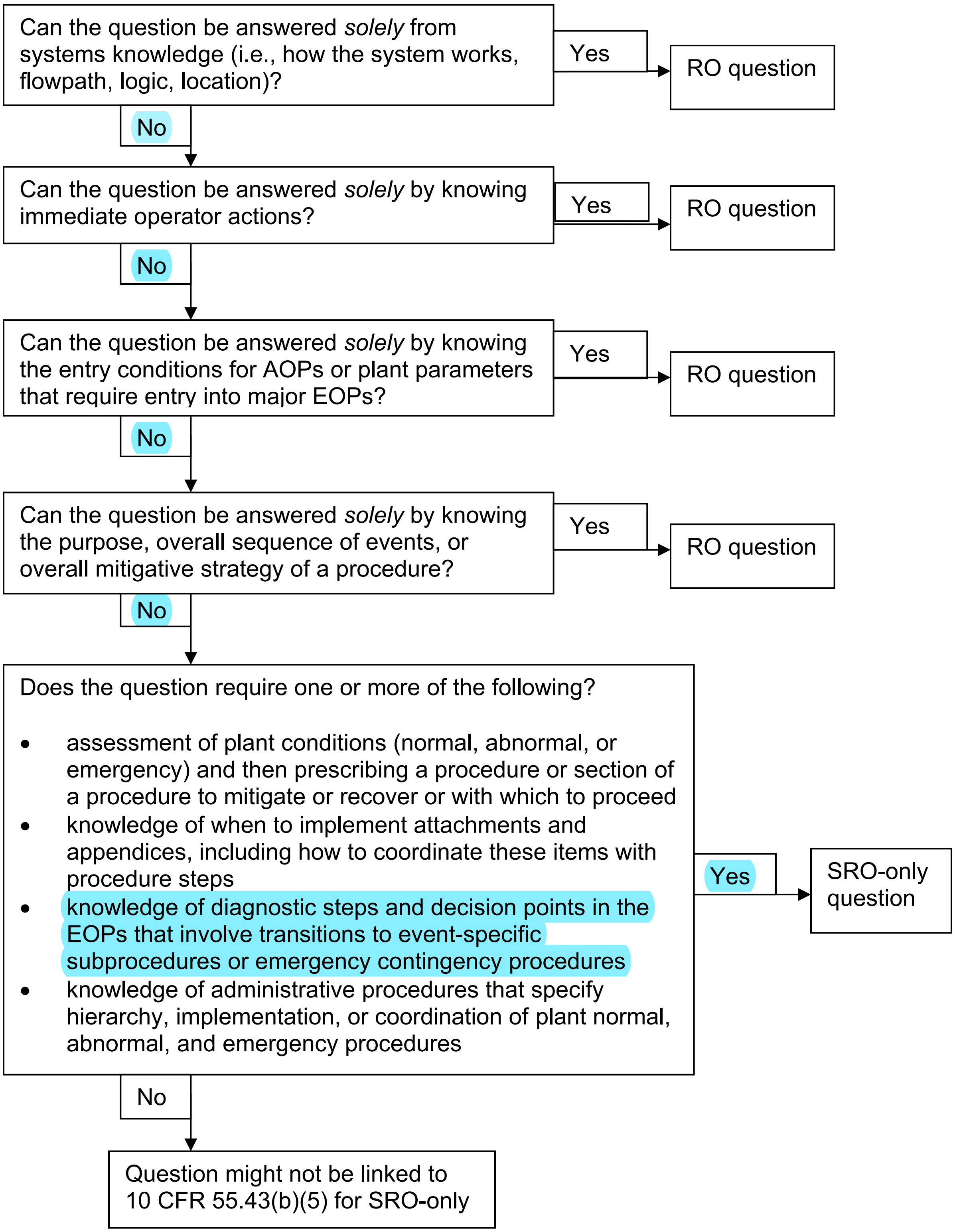
Learning Objective: 5433 -Identify transitions out of and between EOPs

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	N
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

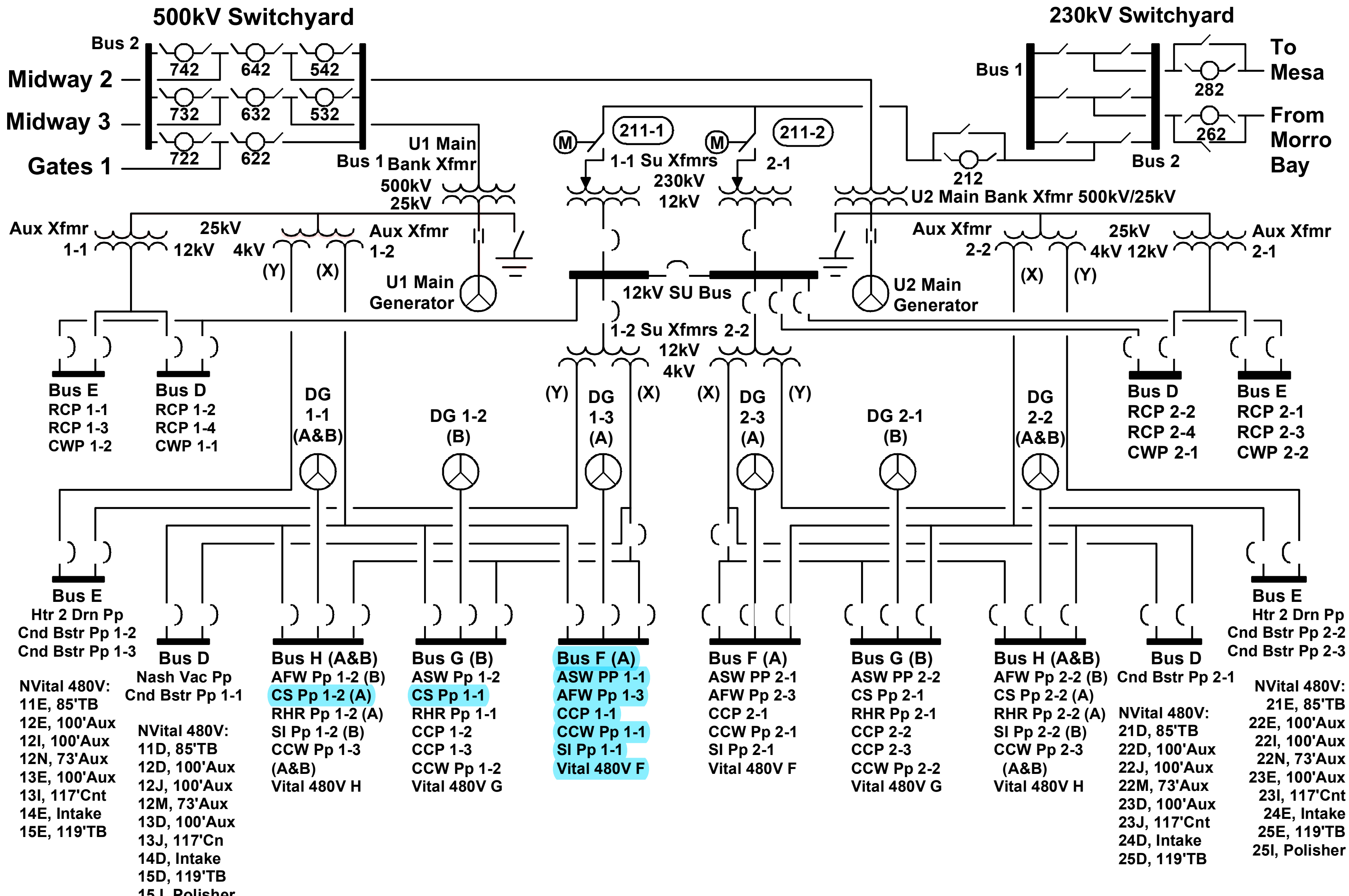
Difficulty: 2.5

Question 10

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Electrical Distribution Overview



Note: OPI.DC38, Att 2, Pg. 2 for information on ECCS/Safe Shutdown Trains

APPENDIX E

ESF AUTO ACTIONS, SECONDARY AND AUXILIARIES STATUS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. NOTIFY Plant Personnel	
a. Check NO personnel in containment	a. Actuate Containment Evacuation Alarm
b. Announce Reactor Trip/SI on PA system	
2. CHECK Main Generator Tripped:	
a. PK14-01, Unit Trip - ON	a. IF Backfeeding, THEN GO TO step 3. IF NOT Backfeeding, THEN Manually initiate a Main Unit Trip.
b. 500KV bkrs - Both Open 1) Green lights - ON OR 2) Turbine speed - LOWERING	b. Open Main Gen Output Bkrs 532 and 632. IF Either breaker will NOT open, THEN Notify GCC Diablo Control (8-449-6717) to isolate Unit 1 at the 500KV switchyard.
c. Exciter Field Breaker - OPEN	c. WHEN Both 500KV bkrs open, THEN Open the Exciter Field Breaker manually or locally.

APPENDIX E

ESF AUTO ACTIONS, SECONDARY AND AUXILIARIES STATUS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. ENSURE Containment Isolation Phase A: a. Phase A portion of Monitor Light Box B • Red Activated Lights - ON • White Status Lights - OFF	Perform the following: • Manually actuate CONTMT ISOL PHASE A, OR • Manually Close the Phase A Isol vlvs with White Status Lights - ON. IF Any flowpath remains un-isolated, THEN Dispatch Operator to locally isolate the flowpath AND Immediately inform the Shift Manager.
4. ENSURE Containment Vent Isol: a. Containment Vent Isol portion of Monitor Light Box B: • Red Activated Lights - ON • White Status Lights - OFF	Perform the following: • Actuate CVI by Manual CONTMT ISOL PHASE A, OR • Manually Close the CVI vlvs with White Status Lights - ON. IF Any flowpath remains un-isolated, THEN Dispatch Operator to locally isolate the flowpath AND Immediately inform the Shift Manager.

APPENDIX E

ESF AUTO ACTIONS, SECONDARY AND AUXILIARIES STATUS

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
5. ENSURE SI Status:	
a. SI portion of Monitor Light Box C: • Red Activated Lights - ON • White Status Lights - OFF	a. Manually actuate SI, OR Start ESF Pps and Align ESF Vlvs with White Status Lights - ON. IF Function is NOT restored, THEN Implement the following as required: • ASW - REFER TO OP AP-10, LOSS OF AUXILIARY SALT WATER • CCW - REFER TO OP AP-11, MALFUNCTION OF COMPONENT COOLING WATER SYSTEM • CCP/CVCS - REFER TO OP AP-17, LOSS OF CHARGING • RHR - REFER TO PK02-17, RHR PUMPS • SI Pps - REFER TO PK02-03, SI PPS TEMP/OC TRIP • CFCU - REFER TO PK01-21, CONTMT FAN CLRS • MD AFW Pps - REFER TO PK09-17, MOTOR DRIVEN AUX FW PP

APPENDIX E (Continued)

ESF AUTO ACTIONS, SECONDARY AND AUXILIARIES STATUS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. ENSURE Feedwater Isolation	
a. Feedwater Isol portion of Monitor Light Box C: • Red Activated Lights - ON • White Status Lights - OFF	a. Manually Close S/G Mn Fdwtr Control vlvs AND Mn Fdwtr Cont Bypass Vlvs with White Status Lights - ON.
b. S/G Lvl portion of Monitor Light Box C: • Red Activated Lights - ON • White Status Lights - OFF	b. Manually Close S/G Mn Fdwtr Isol Vlvs with White Status Lights - ON.
7. CHECK Containment Spray and Phase B Isol - NOT REQUIRED:	
a. Check Phase B portion of Monitor Light Box D: • Red Activated Lights - OFF	a. Perform the following: 1) Ensure Containment Spray initiated. 2) Manually align Phase B Isol Vlvs and Containment Spray components with White Status Lights - ON. 3) Maintain RCP Seal Injection between 8 GPM and 13 GPM. 4) Stop all RCPs.

APPENDIX E (Continued)

ESF AUTO ACTIONS, SECONDARY AND AUXILIARIES STATUS

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
8. CHECK Main Steamline Isol NOT REQUIRED	
a. Check Main Steam Isol portion of Monitor Light Box D: • Red Activated Lights - OFF	a. Ensure White Status Lights - OFF IF Any White Status Lights are ON, THEN Ensure Closed: • MSIVs • MSIV Bypass Valves • S/G Blowdown Valves I.C. IF Any MSIV or MSIV Bypass Vlv will NOT close, THEN Implement Appendix L to locally close valves.
9. CHECK AFW STATUS	
a. MD AFW Pumps - RUNNING • AFW Pump 1-2 • AFW Pump 1-3	a. Start MD AFW Pumps.
b. TD AFW Pump 1-1 - RUNNING IF NEEDED	b. Start TD AFW Pump 1-1, if needed
c. Total AFW flow - GREATER THAN 435 GPM	c. IF S/G NR Level in at least one S/G - GREATER THAN 15% [25%], THEN GO TO step 10 (page 25). IF S/G NR Level - LESS THAN 15% [25%] in all S/Gs, THEN Manually start pumps and align valves as necessary to restore AFW flow.

APPENDIX E (Continued)

ESF AUTO ACTIONS, SECONDARY AND AUXILIARIES STATUS

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
10. CHECK ECCS Flows:	
a. Charging Injection Flow, FI-917 - GREATER THAN 100 GPM	a. Perform the following: 1) Ensure both ECCS CCPs Running. 2) IF ECCS CCPs running at LESS THAN minimum recirc flow, 48 AMPS, THEN Ensure ECCS valve alignment: a) Check monitor light Box A and C. b) Align valves with White Status Lights - ON. c) Ensure manual flowpath valves aligned.
b. Check CCP 1-3 - OFF	b. IF Both ECCS CCPs are in service, THEN Shutdown CCP 1-3.
c. RCS WR Pressure - LESS THAN 1650 PSIG	c. GO TO step 11 (page 27).
d. SI Pp Flow, FI-918/922 - GREATER THAN 100 GPM	d. Perform the following: 1) Ensure both SI Pps running, 2) IF SI Pp running LESS THAN minimum recirc flow, 28 AMPS, THEN Check ECCS valve alignment: a) Check Monitor Light Box A and C. b) Align valves with White Status Lights - ON. c) Ensure manual flowpath valves aligned.

THIS STEP CONTINUED ON NEXT PAGE

APPENDIX E (Continued)

ESF AUTO ACTIONS, SECONDARY AND AUXILIARIES STATUS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. CHECK ECCS Flows: (Continued)	
e. RCS WR Pressure - LESS THAN 300 PSIG	e. GO TO step 11 (page 27).
f. RHR Pp flow, FI-970A/917A - GREATER THAN 100 GPM	f. Perform the following: 1) Ensure both RHR Pps running. 2) IF RHR Pps running at LESS THAN minimum recirc flow, 28 AMPS, THEN Check ECCS valve alignment: a) Check Monitor Light Box A and C. b) Align valves with White Status Lights - ON. c) Ensure manual flowpath valves aligned.

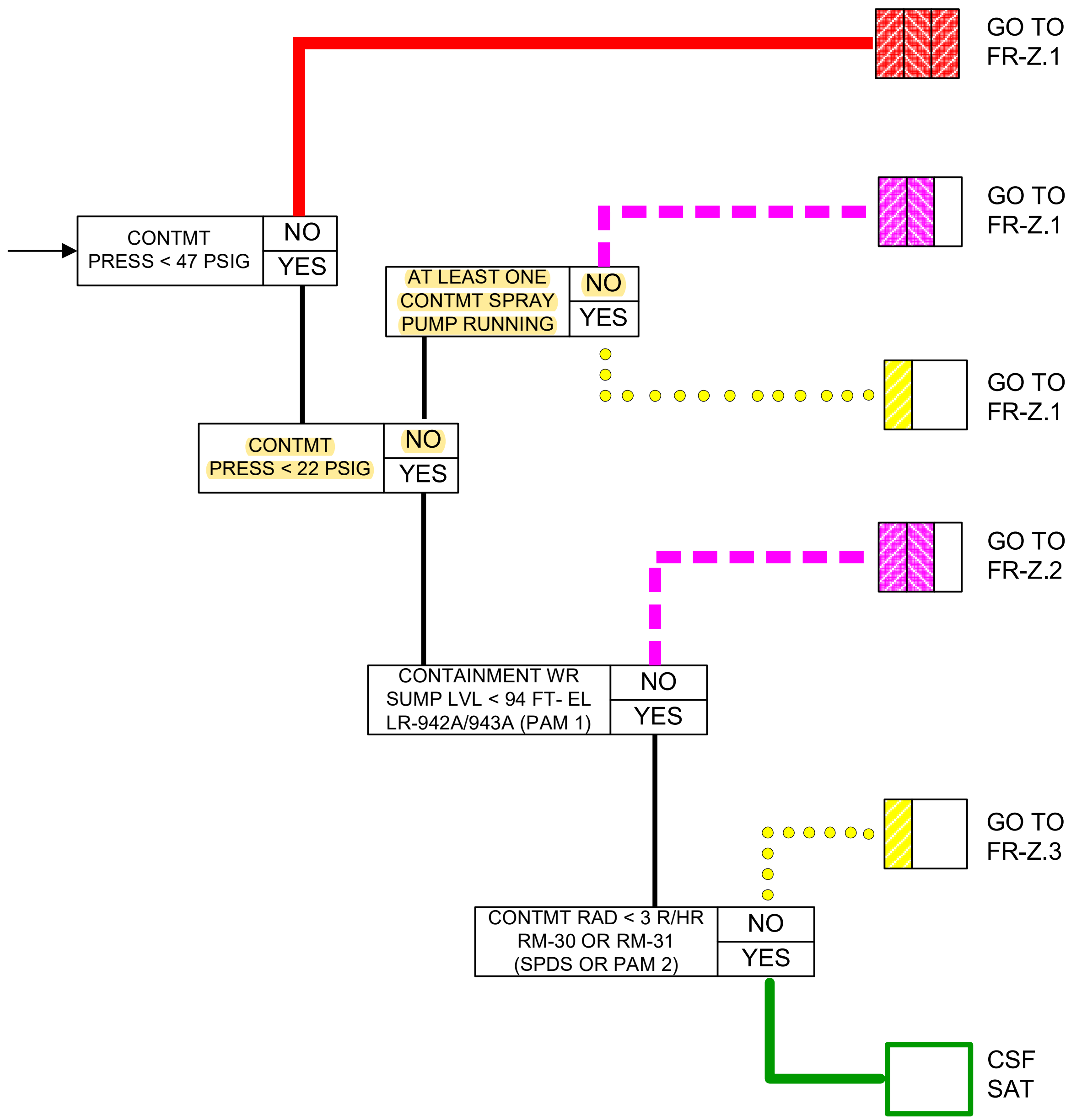
APPENDIX E (Continued)

ESF AUTO ACTIONS, SECONDARY AND AUXILIARIES STATUS

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
11. REPORT to SFM:	
a. ESF and AFW status	
b. Any abnormalities	
12. Check Excess Letdown - NOT IN SERVICE PRIOR TO TRIP	Place 8166 AND 8167 control switches in CLOSE position (VB2).
13. CHECK Secondary System Status:	
a. Check Mn Fdwtr Pps Tripped • Mn Fdwtr Pps 1-1 and 1-2 Green Trip Light - ON	a. Manually or locally trip Mn Fdwtr Pps.
b. Stop ALL but one Condensate and Booster Pp Set	
c. Place LCV-12 in CONT ONLY	

THIS STEP CONTINUED ON NEXT PAGE

F-0.5 CONTAINMENT



1.0 **CONTAINMENT SPRAY INITIATION CRITERIA**

IF	Contmt Pressure is GREATER THAN 22 PSIG,
THEN	Initiate Contmt Spray.

2.0 **COLD LEG RECIRCULATION SWITCHOVER CRITERION**

IF	RWST Level lowers to LESS THAN 33%,
THEN	IMPLEMENT EOP E-1.3, TRANSFER TO COLD LEG RECIRCULATION.

3.0 **RESTART SAFEGUARDS EQUIPMENT AFTER LOSS OF OFFSITE POWER**

IF	Offsite Power is lost AFTER SI RESET,
THEN	Restart Safeguards equipment as necessary (REFER TO APPENDIX A for guidance).
IF	In recirculation mode,
THEN	ECCS CCPs should be held in STOP/RESET until RHR is in service.

4.0 **ESTABLISHING FEED TO DRY STEAM GENERATOR CRITERIA**

A "Dry" S/G is a S/G with WR level LESS THAN 10% [18%]. Feeding a Dry S/G should be performed only when NO other intact S/G is available, unless depressurizing this S/G for low pressure feedwater injection.

1)	IF	Bleed and feed has been initiated,
	AND	Core exit T/C temperatures are rising,
	THEN	Feed ONE Dry S/G at MAXIMUM rate until WR level is ≥ 10% [18%].
2)	IF	Bleed and feed has been initiated,
	AND	Core exit T/C temperatures are stable or lowering,
	THEN	Feed ONE Dry S/G at 25-100 GPM until WR level is ≥ 10% [18%].
3)	WHEN	WR level is ≥ 10% [18%],
	THEN	Adjust feed rate as necessary to restore heat sink,
	AND	Check for SGTR. Use another S/G if a SGTR exists.

5.0 **BLEED AND FEED CRITERIA**

IF	WR S/G Level in any THREE (3) S/Gs LESS THAN 18% [26%],
	AND
	ALL NR S/G Levels are LESS THAN 15% [25%],
THEN	STOP ALL RCPS AND Initiate Bleed and Feed, steps 12 through 18.

6.0 **SECONDARY INTEGRITY CRITERIA**

IF	Any S/G Pressure is lowering in an Uncontrolled manner or has completely depressurized,
	AND has NOT been isolated, unless it is needed for cooldown,
THEN	REFER TO EOP E-2, FAULTED STEAM GENERATOR ISOLATION.

7.0 **AFW SUPPLY SWITCHOVER CRITERION**

IF	CST Level lowers to LESS THAN 10%,
THEN	IMPLEMENT OP D-1:V, ALTERNATE AFW SUPPLIES.

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	2
005 A2.05 - Ability to (a) predict the impacts of the following on the Residual Heat Removal System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: RHR heat exchanger malfunction	**Group #**	1
	K/A #	005 A2.05
	Rating	3.5

Question 11

Unit 1 RCS temperature is 250°F.

A tube leak develops in the in service RHR heat exchanger.

The Shift Foreman should enter:

A. OP AP-11, Section B - CCW System Inleakage

B. OP AP-11, Section C - CCW System Outleakage

C. OP AP SD-4, Loss of Component Cooling Water

D. OP AP SD-5, Loss of Residual Heat Removal

Proposed Answer: A. OP AP-11, Section B - CCW System Inleakage

Explanation:

A. Correct. RHR pressure is higher than CCW pressure. Because the plant is in MODE 4 (RCS temperature is 250°F), OP AP-11 is the applicable procedure and section B, for system in leakage is the correct procedure selection.
B. Incorrect. Plausible, OP AP-11 is correct, however, RHR will flow into CCW not CCW into RHR. Plausible if the pressure was thought to be higher in CCW.
C. Incorrect. Plausible as OP AP SD-4 would eventually direct isolation however, it is not applicable in MODE 4 and applicable only in MODE 5 and 6.
D. Incorrect. Plausible as OP AP SD-5 is not applicable in MODE 4 but would address the leak.

Technical References: OP AP-11, OP AP SD-4, OP AP SD-5

References to be provided to applicants during exam: None

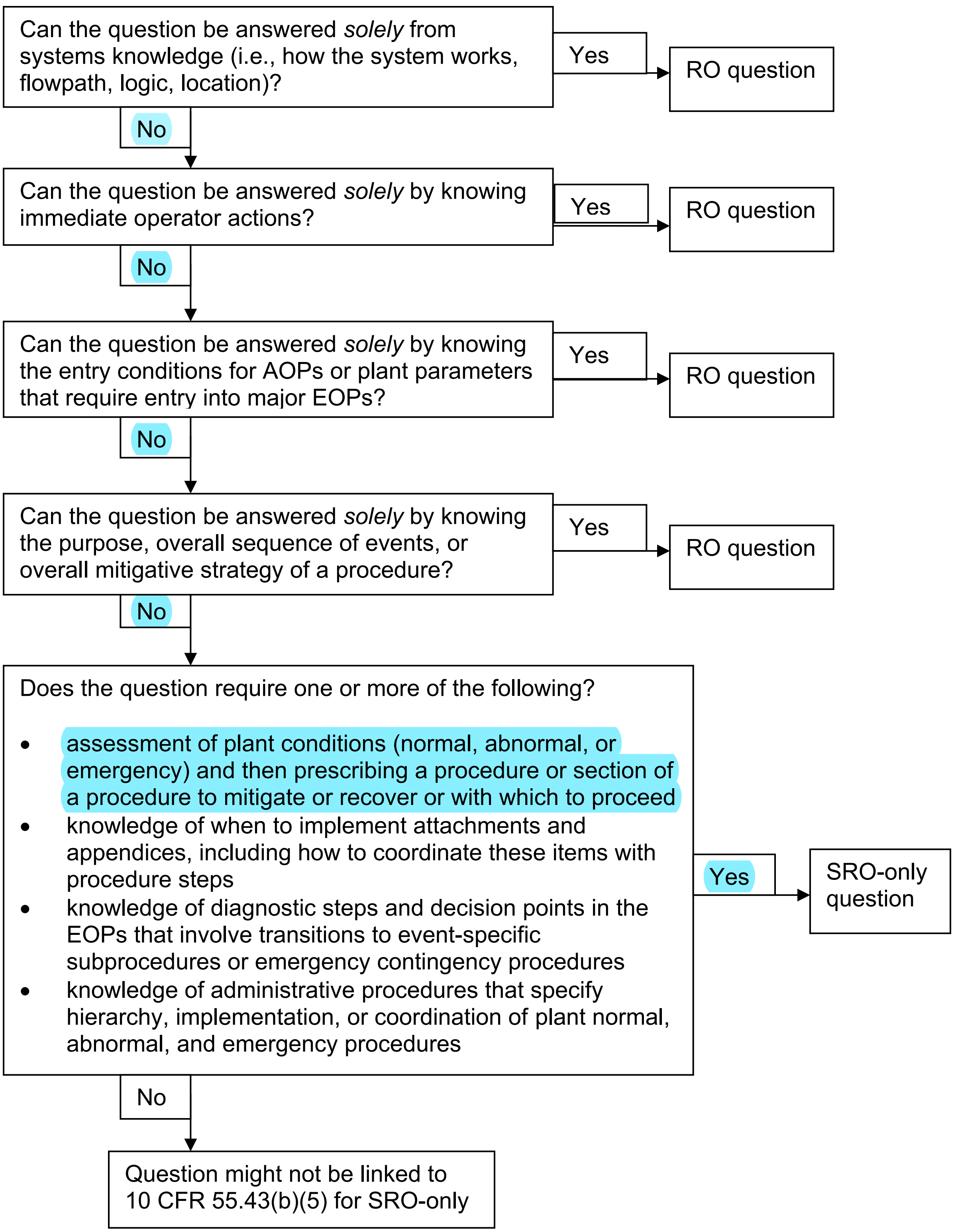
Learning Objective: 3478 - Given initial conditions, assumptions, and symptoms, determine the correct abnormal operating procedure to be used to mitigate an operational event

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	N
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 3

Question 11

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Malfunction of Component Cooling Water System

06/03/19
Effective Date

QUALITY RELATED

1. SCOPE

1.1 This procedure covers Component Cooling Water (CCW) System leakage or loss of cooling to various vital components while in MODES 1-4. If in MODE 5 or 6, OP AP SD-4, LOSS OF COMPONENT COOLING WATER, should be used if Decay Heat Removal is threatened.

1.2 Prompt corrective action is vital to prevent complete deterioration of the system. The primary action is to isolate the defective component or section and terminate the leakage.[T36122]

2. SYMPTOMS OR ENTRY CONDITIONS

See Appropriate Section

Loss of Component Cooling Water

04/25/16
Effective Date

QUALITY RELATED

1. SCOPE

1.1 This procedure is used in Modes 5, 6 or core offload when Decay Heat Removal is threatened due to a failure of the CCW System. This failure may have been due to insufficient flow, CCW System IN or OUT leakage, inadequate makeup or high system temperature.

1.1.1 Step 1 selects the correct part of the procedure to use to resolve the current problem.

1.1.2 Steps 2 thru 7 are to be used if the CCW temperature is high or no ASW Pumps are available.

a. A second train of ASW is placed in service if available.

b. ASW flow is verified to be adequate, if not a transition to OP AP SD-3, "Loss of Auxiliary Salt Water" is made, unless this procedure was entered from OP AP SD-3 or it has been determined that ASW Pumps are not capable of being placed in service, then temporary cooling is aligned to the CCW Heat Exchangers.

c. Heat loads into the CCW System are then reduced and alternative methods of heat removal from the CCW System are attempted.

1.1.3 Steps 7 thru 10 are to be used if CCW System flow is low.

a. If no CCW pumps can be started or no ASW Pumps are available then the heat inputs to the CCW System are minimized and a transition is made to OP AP SD-0.

b. If only one CCW pump can be run then CCW flow is reduced to within the capability of one CCW pump.

1.1.4 Steps 11 thru 14 are to be used if there is a leak OUT of the CCW System.

a. Makeup capabilities are verified.

b. The leaking CCW component is identified and isolated.

Loss of Residual Heat Removal

04/01/14
Effective Date

QUALITY RELATED

1. SCOPE

1.1 This procedure is used in Modes 5 and 6 when RHR system flow is lost. It provides the actions necessary to regain flow and actions to be taken if RHR flow cannot be regained.

1.2 This procedure is NOT to be used if an inventory problem has caused, or has the potential to cause, the loss of the RHR system. Such cases are covered by OP AP SD-2, "Loss of RCS Inventory."

2. SYMPTOMS OR ENTRY CONDITIONS

2.1 Rising reactor cooling temperature indication on core exit thermocouples and/or loop wide range RTDs.

2.2 Rising RHR Heat Exchanger outlet temperatures.

2.3 Loss of flow indication on FI-970 A and B and/or FI-971 A and B.

2.4 Possible Annunciator Alarms:

2.4.1 RHR SYSTEM (PK02-16)

a. RHR Pp _____ Discharge pressure Hi

b. RHR Pp _____ Discharge flow Low

2.4.2 RHR PUMPS (PK02-17)

a. RHR pump trouble alarms

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	2
013 G2.2.42 – ESFAS: Ability to recognize system parameters that are entry-level conditions for TS	**Group #**	1
	K/A #	013 G2.2.42
	Rating	4.6

Question 12

GIVEN:

- Unit 1 is at 100% power.
- 73 hours ago, Steam Generator 1-1 pressure channel PT-514 in Protection Set 1 - Rack 3, was inoperable and the actions to satisfy LCO 3.3.2 Condition D have been completed

The Shift Foreman has just declared Steam Generator 1-4 pressure channel PT-544 in Protection Set 1 - Rack 3 inoperable due to a failed surveillance.

What action should be taken by the Shift Foreman?

A. Enter LCO 3.0.3 and be in MODE 3 within the next 7 hours

B. Be in MODE 3 in a maximum of 30 hours

C. Be in MODE 3 in a maximum of 78 hours

D. Trip the applicable bistables within 72 hours

Proposed Answer: D. Trip the applicable bistables within 72 hours

Explanation:

A. Incorrect. Plausible if there is a lack of familiarity with the bases statement about the note, then it could be thought that separate function(s) is for different functions, ie failed NI and failed level transmitter, both of which use ACTION D.

Additionally, if the failure was a channel on the same steam generator, LCO action would be required.

B. Incorrect. Plausible, as 30 hours is a typical time to MODE 3 in many ACTION such as C.2

C. Incorrect. Plausible if its assumed the bistables cannot be tripped without causing a reactor trip and so the shutdown action of D.2 applies.

D. Correct. Per the note, separate entry is allowed for each function. The Bases for LCO 3.3.2 states “In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. *When the Required Channels in Table 3.3.2-1 are specified (e.g., on a per steam line, per loop, per SG), then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.*

Technical References: LCO 3.3.2 and B3.3.2

References to be provided to applicants during exam: first 3 pages of LCO 3.3.2

Learning Objective: 9697C - Apply TS 3.3 Technical Specification LCOs.

Question Source:	Bank # L181 Question 82	Y
(note changes; attach parent)	Modified	
	New	
	Past NRC Exam DCPP 03/2020	Y

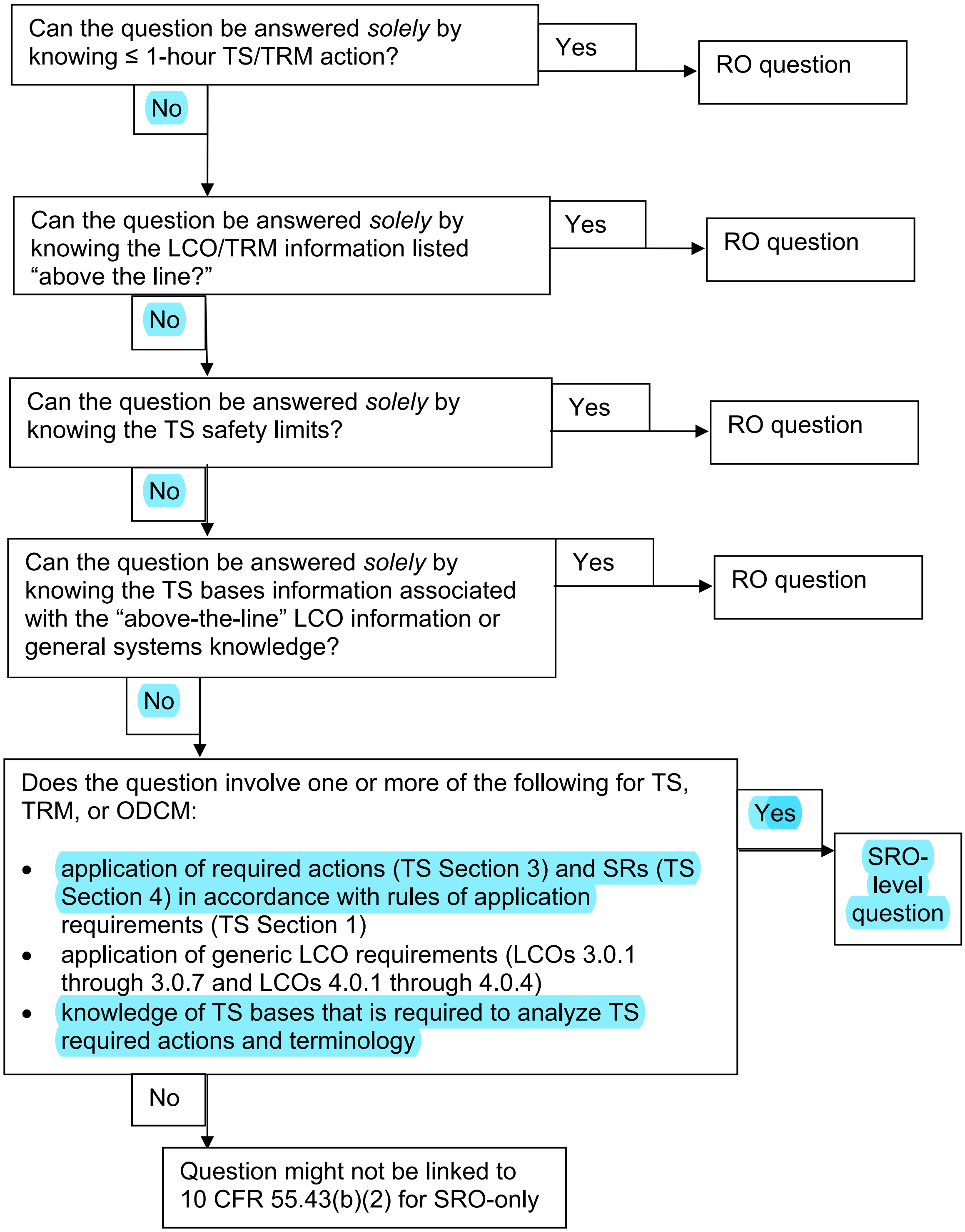
Last Two NRC Exams	Y

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content: 55.43.2
Difficulty: 2.7

Question 12

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



c. *Facility Licensee Procedures Required to Obtain Authority for Design and Operating Changes in the Facility [10 CFR 55.43(b)(3)]*

Some examples of SRO-only examination items for this topic include the following:

Examination Outline Cross-Reference		**Level**	SRO
		Tier #	2
035 A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the S/GS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Pressure/level transmitter failure		**Group #**	2
		K/A #	035 A2.03
		Rating	3.6

Question 82 PARENT QUESTION

Unit 1 is at 100% power.

73 hours ago, Steam Generator 1-1 pressure channel PT-514 in Protection Set 1 - Rack 3, was inoperable and the actions to satisfy LCO 3.3.2 Condition D have been completed.

The Shift Foreman has just declared Steam Generator 1-4 pressure channel PT-544 in Protection Set 1 - Rack 3 inoperable due to a failed surveillance.

What action should be taken by the Shift Foreman?

A. Enter LCO 3.0.3

B. Be in MODE 3 in a maximum of 30 hours

C. Be in MODE 3 in a maximum of 78 hours

D. Trip the applicable bistables within 72 hours

Proposed Answer: D. Trip the applicable bistables within 72 hours

Explanation:

A. Incorrect. If there is a lack of familiarity with the bases statement about the note, then it could be thought that separate function(s) is for different functions, ie failed NI and failed level transmitter, both of which use ACTION D.
B. Incorrect. 30 hours is a typical time to MODE 3 in many ACTION such as C.2
C. Incorrect. If its assumed the bistables can't be tripped and so the shutdown action of D.2 applies.
D. Correct. Per the note, separate entry is allowed for each function. The Bases for LCO 3.3.2 states "In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. *When the Required Channels in Table 3.3.2-1 are specified (e.g., on a per steam line, per loop, per SG), then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.*

Technical References: LCO 3.3.2 and B3.3.2

References to be provided to applicants during exam: first 3 pages of LCO 3.3.2

Learning Objective: 9697C - Apply TS 3.3 Technical Specification LCOs.

Question Source:	Bank #
(note changes; attach parent)	Modified Bank #

3.3 INSTRUMENTATION

3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.2 The ESFAS instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2-1.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels or trains inoperable.	A.1 Enter the Condition referenced in Table 3.3.2-1 for the channel(s) or train(s).	Immediately
B. One channel or train inoperable.	B.1 Restore channel or train to OPERABLE status.	48 hours
	<u>OR</u>	
	B.2.1 Be in MODE 3.	54 hours
	<u>AND</u>	
	B.2.2 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------ Be in MODE 4.	60 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One train inoperable.	----------------NOTE----------------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. --	
	C.1 Restore train to OPERABLE status.	24 hours
	OR	
	C.2.1 Be in MODE 3.	30 hours
	AND	
	C.2.2 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------ Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One channel inoperable.	------------------NOTE------------------ For function 1.d, the inoperable channel and/or one additional channel may be surveillance tested with one channel in bypass and one channel in trip for up to 12 hours, or both the inoperable and the additional channel may be surveillance tested in bypass for up to 12 hours. For functions 1.e(1), 4.d(1), 4.d(2), and 6.d(1), the inoperable channel and/or one additional channel may be surveillance tested with one channel in bypass and one channel in trip for up to 12 hours. This note is not intended to allow simultaneous testing of coincident channels on a routine basis. --	
	D.1 Place channel in trip.	72 hours
	OR	
	D.2.1 Be in MODE 3.	78 hours
	AND	
	D.2.2 Be in MODE 4.	84 hours

(continued)

Table 3.3.2-1 (page 1 of 7)
Engineered Safety feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL[a*] TRIP SETPOINT
1. Safety Injection						
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.8	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Containment Pressure-High	1,2,3,4	3	O	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 3.12 psig	3.0 psig
d. Pressurizer Pressure-Low	1,2,3[b]	4	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 1847.5 psig	1850 psig
e. Steam Line Pressure						
(1) Low	1,2,3[b]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 597.6[c] psig	600[c] psig
(2) Not used						
f. Not used						
g. Not used						

(continued)

(a) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.

(b) Above the P-11 (Pressurizer Pressure) interlock and below the P-11 interlock unless the Function is blocked.

(c) Time constants used in the lead/lag compensator are t_1 = 50 seconds and t_2 = 5 seconds.

Table 3.3.2-1 (page 4 of 7)
Engineered Safety feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL [(a)] TRIP SETPOINT
4. Steam Line Isolation (continued)						
d. Steam Line Pressure						
(1) Low	1,2[(i)], 3[(b)(i)]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 597.6[(c)] psig	600[(c)] psig
(2) Negative Rate-High	3[(g)(i)]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 102.4 [(h)] psi/sec	100 [(h)] psi/sec
e. Not used.						
f. Not used						
g. Not used						
h. Not used						
5. Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	1,2[(j)]	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA

(continued)

(a) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.
(b) Above the P-11 (Pressurizer Pressure) interlock and below the P-11 interlock unless the Function is blocked.
(c) Time constants used in the lead/lag compensator are t_1 = 50 seconds and t_2 =5 seconds
(g) Below the P-11 (Pressurizer Pressure). However, may be blocked below P-11 when Safety Injection on Steam Line Pressure-Low is not blocked.
(h) Time constant utilized in the rate/lag compensator are t_3 = 50 sec and t_4 = 50 sec.
(i) Except when all MSIVs are closed and de-activated.
(j) Except when all MFIVs, MFRVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve.

Table 3.3.2-1 (page 5 of 7)
Engineered Safety feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL [(a)] TRIP SETPOINT
5. Feedwater Isolation (continued)						
b. SG Water Level-High High (P-14)	1,2[(j)]	3 per SG	J	SR 3.3.2.1 SR 3.3.2.5[(d)(e)] SR 3.3.2.9[(d)(e)] SR 3.3.2.10	≤ 90.2%	90.0%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
6. Auxiliary Feedwater						
a. Manual	1,2,3	1 sw/pp	N	SR 3.3.2.13	NA	NA
b. Automatic Actuation Logic and Actuation Relays (Solid State Protection System)	1,2,3	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Not used						
d.1 SG Water Level-Low Low	1,2,3	3 per SG	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 14.8%	15.0%

(continued)

(a) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.

(j) Except when all MFIVs, MFRVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve.

(d) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. Footnote (a) does not apply to this function.

(e) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in the Equipment Control Guidelines. Footnote (a) does not apply to this function.

BASES

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.2-1.

With an ESFAS Instrumentation Channel or Interlock Trip Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Values column of Table 3.3.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument Loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. When the Required Channels in Table 3.3.2-1 are specified (e.g., on a per steam line, per loop, per SG, etc., basis), then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.

When the number of inoperable channels in a trip function exceed those specified in one or other related Conditions associated with a trip function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

A.1

Condition A applies to all ESFAS protection functions.

Condition A addresses the situation where one or more channels or trains for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.2-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

(continued)

BASES

ACTIONS (continued)

D.1, D.2.1, and D.2.2

Condition D applies to:

- SI - Pressurizer Pressure — Low;
- SI - Steam Line Pressure — Low;
- Steam Line Isolation - Steam Line Pressure — Negative Rate — High;
- Steam Line Isolation - Steam Line Pressure — Low; and
- Auxiliary Feedwater - SG Water level — Low Low;

If one channel is inoperable, 72 hours are allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Generally this Condition applies to functions that operate on two-out-of-three logic (excluding pressurizer pressure - low which is two-out-of-four due to its control input function). Therefore, failure of one channel places the Function in a two-out-of-two configuration. The inoperable channel must be tripped to place the Function in a one-out-of-two configuration that satisfies redundancy requirements. Since pressurizer pressure is used for control and SSPS input, its coincidence is two-out-of-four to provide to required reliability and redundancy. Failure of one channel places the function in a two-out-of-three configuration. The inoperable channel must be placed in the tripped condition to place the Function in a one-out-of-three configuration that satisfies the reliability and redundancy requirements.

(continued)

BASES

ACTIONS

D.1, D.2.1, and D.2.2 (continued)

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 72 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note for Function 1.d that allows the inoperable channel and/or one additional channel to be tested with one channel in bypass and one channel in trip, or with both the inoperable and the additional channel in bypass for up to 12 hours for surveillance testing. For Functions 1.e, 4.d(1), 4.d(2) and 6.d(1), the Note allows the inoperable channel and/or one additional channel to be tested with one channel in bypass and one channel in trip for up to 12 hours for surveillance testing. Function 1.d is a two-out-of-four trip logic and Functions 1.e, 4.d(1), 4.d(2) and 6.d(1) are two-out-of-three logic actuation logics. The allowed testing configurations provide flexibility for testing, while assuring that during testing no configuration will cause an inadvertent actuation of the function or keep a valid signal from actuating the function as it was designed. The 72 hours allowed to restore the channel to OPERABLE status or to place the inoperable channel in the tripped condition, and the 12 hours allowed for testing, are justified in Reference 17. This note is not intended to allow simultaneous testing of coincident channels on a routine basis.

E.1, E.2.1, and E.2.2

Condition E applies to:

- Steam Line Isolation - Containment Pressure - High-high

This signal does not input to a control function. Thus, two-out-of-three logic is necessary to meet acceptable protective requirements. However, a two-out-of-three design would require tripping a failed channel. This is undesirable because a single failure of the Containment Pressure input would then cause spurious containment spray initiation. Spurious spray actuation is undesirable because of the cleanup problems presented. Therefore, these channels are designed with two-out-of-four logic so that a failed channel may be bypassed rather than tripped. Note that one channel may be bypassed and still satisfy the single failure criterion. Furthermore, with one channel bypassed, a single instrumentation channel failure will not spuriously initiate containment spray.

(continued)

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	2
059 A2.04 Ability to (a) predict the impacts of the following on the Main Feedwater System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Malfunctioning steam dump	**Group #**	1
	K/A #	059 A2.04
	Rating	3.7

Question 13

GIVEN:

- Unit 1 has tripped and the crew has transitioned to EOP E-0.1, Reactor Trip Response
- RCS temperature is 542°F and lowering slowly
- RCS pressure is 1950 psig and lowering slowly
- Steam dumps are in Steam Pressure mode
- Pressurizer level is 15% and lowering slowly

The BOPCO reports that two condenser steam dump valves are open.

In accordance with EOP E-0.1, what actions should be taken by the Shift Foreman?

A. Direct the crew to actuate safety injection and return to EOP E-0, Reactor Trip or Safety Injection, step 1, ENSURE Reactor Trip.

B. Direct the crew to actuate safety injection and return to EOP E-0, Reactor Trip or Safety Injection, step 4, CHECK SI - ACTUATED.

C. Direct the crew to close the MSIVs and adjust the 10% steam dumps to control at approximately 1005 psig.

D. Direct the crew to close the MSIVs and adjust the 10% steam dumps to control at approximately 1040 psig.

Proposed Answer: C. Direct the crew to close the MSIVs and adjust the 10% steam dumps to control at approximately 1005 psig.

Explanation:

The SRO will assess conditions against criteria in EOP E-0.1 and determine if SI must be initiated. Once determining SI is not required, decide what action should be taken with the steam dump per procedure to stop the cooldown.

A. Incorrect. Plausible - if pressurizer level was less than 6% or RCS subcooling lower (less than 20°F, a transition back to EOP E-0 and SI actuation would be required. Also, if its thought there is a steam break the action of actuating SI and taking procedure flowpath to get to EOP E-2 is plausible.

B. Incorrect. Plausible if pressurizer level was less than 6% or RCS subcooling lower (less than 20°F, a transition back to EOP E-0 and SI actuation would be required. Additionally, plausible because many EOPs or FRGs direct returning to procedure and step in effect. The transition to EOP E-0.1 is from step 4 of EOP E-0. 15% in pressurizer is plausible as it is the setpoint for adequate steam generator level, could be thought that it is the level for actuating SI.

C. Correct. Per the RNO (step 1), action for a continued cooldown is to close the MSIVs and adjust 10% steam dumps for no load RCS temperature, 547°F, or 1005 psig.

D. Incorrect. Plausible as this is the setpoint for adjusting steam dumps for a steam generator tube rupture in EOP E-3.

Technical References: EOP E-0.1 step 1 and foldout page

References to be provided to applicants during exam: None

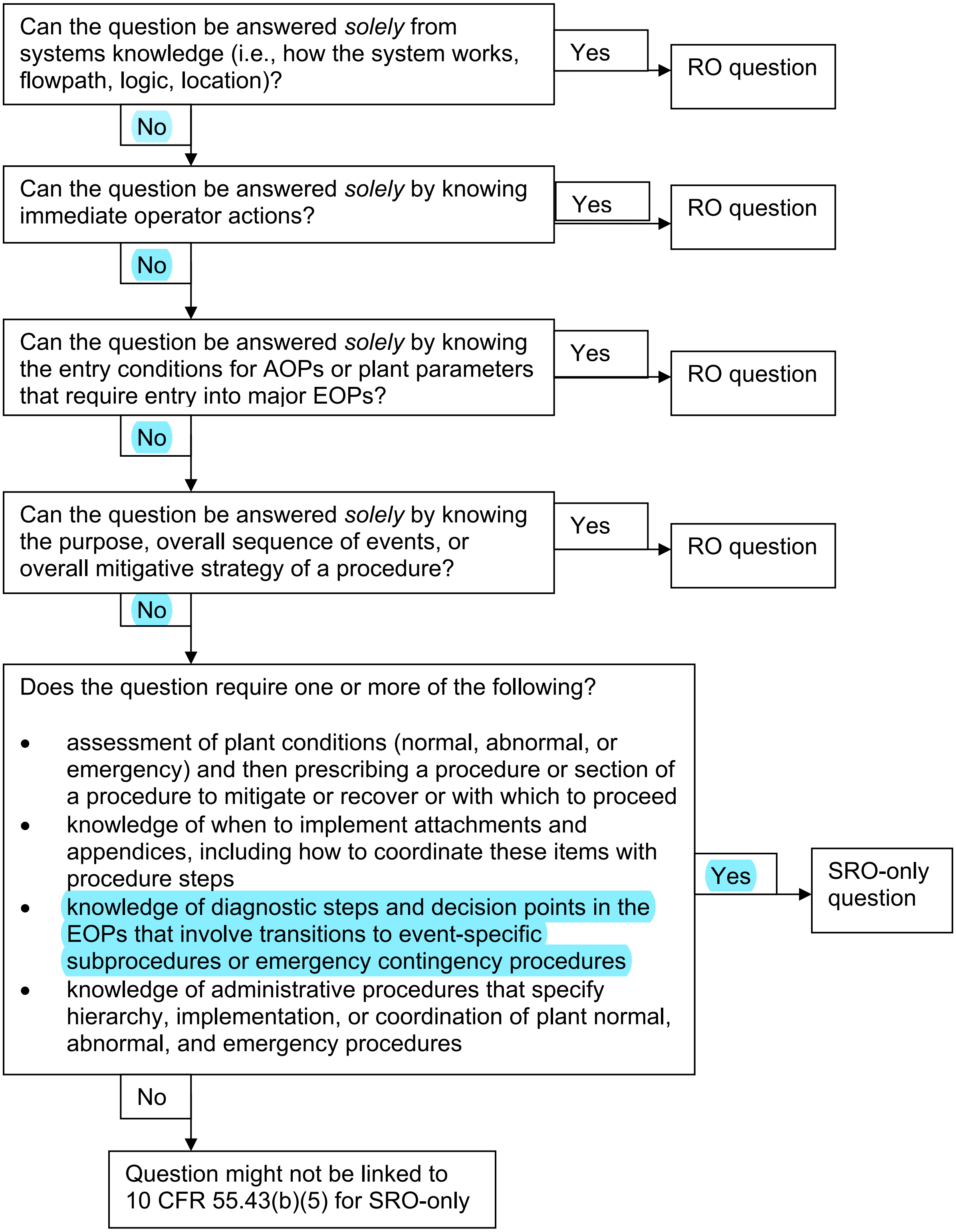
Learning Objective: 3552 Given initial conditions, assumptions, and symptoms, determine the correct Emergency Operating Procedure to be used to mitigate an operational event

Question Source:	Bank #88 L091C	Y
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 03/2012	Y
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 3

Question 13

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Examination Outline Cross-Reference	Level	SRO
039 A2.04 Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Malfunctioning steam dump	**Tier #**	2
	Group #	1
	K/A #	039 A2.04
	Rating	3.7

Question SRO 13 (88) PARENT QUESTION

GIVEN:

- Unit 2 has tripped and the crew has just transitioned to E-0.1, Reactor Trip Response
- RCS temperature is 542°F and lowering
- RCS pressure is 1950 psig and lowering
- Pressurizer level is 12% and lowering
- The BOPCO reports that two condenser steam dump valves are open

In accordance with E-0.1, which of the following actions should be taken by the Shift Foreman?

A. Direct the crew to initiate safety injection and return to E-0, Reactor Trip or Safety Injection, step 1.

B. Direct the crew to initiate safety injection and return to E-0, Reactor Trip or Safety Injection, step 4.

C. Direct the crew to close the MSIVs and adjust the 10% steam dumps to control at 1040 psig (8.67 turns).

D. Direct the crew to close the MSIVs and adjust the 10% steam dumps to control at less than or equal to 1005 psig (8.38 turns).

Proposed Answer: D. Direct the crew to close the MSIVs and adjust the 10% steam dumps to control at less than or equal to 1005 psig (8.38 turns).

Explanation:

A. Incorrect. Subcooling, pressure and pressurizer level are all above the points requiring SI actuation.
B. Incorrect. If SI was required, the proper transition would be to return to step 1 of E-0, not the point of transitioning out of E-0.
C. Incorrect. 1040 psig is used in E-3 for the ruptured steam generator.
D. Correct. The unit is below P-11 and the dumps should be closed. The MSIVs (and bypass valves) are closed to stop the cooldown.

Technical References: E-0.1, steam tables

References to be provided to applicants during exam: None

Learning Objective: 3552 Given initial conditions, assumptions, and symptoms, determine the correct Emergency Operating Procedure to be used to mitigate an operational event

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.	

Lowered pressurizer level to be closer to manual SI actuation setpoint. Added Foldout page to question reference and "IAW E-0.1" to question

ACTION/EXPECTED RESPONSE	**RESPONSE NOT OBTAINED**
1. CHECK RCS Temperature Response – STABLE OR TRENDING TO 547°F (Continued)	
b. Check RCS temperature response - NORMAL • Average Temperature if any RCP is running OR • Cold Leg Temperature if NO RCP is running	b. PERFORM the following: IF RCS Temperature is LESS THAN 547°F AND LOWERING, THEN 1) Ensure ALL Steam Dump Valves closed. 2) IF MSIVs are open AND Condenser is available (C-9 in, PK08-14), THEN Transfer condenser steam dumps to Pressure Control Mode by performing the following: a) Transfer HC-507 to manual. b) Ensure demand is 0%. c) Place Steam Dumps in Steam Pressure Mode. d) Adjust steam dump controller, HC-507, as needed to maintain S/G pressure LESS THAN OR EQUAL TO 1005 PSIG (83.8% setpoint). e) Place HC-507 in Auto.

THIS STEP CONTINUED ON NEXT PAGE

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

1. CHECK RCS Temperature Response – STABLE OR TRENDING TO 547°F (Continued)

3) Ensure S/G Blowdown Isol Valves OC – CLOSED.

S/G	Bldn Isol Valve OC	Bldn Sample Isol Valve
1	FCV-151	FCV-250
2	FCV-154	FCV-248
3	FCV-157	FCV-246
4	FCV-160	FCV-244

4) Ensure MSRs Reset (MSR Screen on the Turbine Control Panel).

5) IF Cooldown continues,
THEN Control Total Feed Flow. Maintain total feed flow GREATER THAN 435 GPM until NR level is GREATER THAN 15% [25%] in at least one S/G.

6) IF Cooldown continues,
THEN
a) Close MSIVs and MSIV Bypass Valves.
b) Adjust 10% steam dump controllers as needed to maintain S/G pressure LESS THAN OR EQUAL TO 1005 PSIG (83.8% setpoint).

7) IF Cooldown continues
AND is UNCONTROLLED,
THEN IMPLEMENT OP AP-6 EMERGENCY BORATION.

8) GO TO step 2 (page 6).

THIS STEP CONTINUED ON NEXT PAGE

1.0 **RCP TRIP CRITERIA**

Phase B Isolation has Actuated,

1) Trip ALL RCPs within 5 minutes.

2) Maintain RCP Seal Injection between 8 GPM and 13 GPM, throttling FCV-128.

RCS WR Pressure LESS THAN 1300 PSIG,

1) Ensure either one SI Pp OR one ECCS CCP running, delivering flow.

2) Stop ALL RCPs.

3) Maintain RCP Seal Injection between 8 GPM and 13 GPM, throttling FCV-128.

2.0 **SI ACTUATION CRITERIA**

IF EITHER condition listed below occurs:

- RCS Subcooling based on Core Exit T/Cs - LESS THAN 20°F
- PZR Level - CANNOT BE MAINTAINED GREATER THAN 6%

THEN Actuate SI AND GO TO EOP E-0, REACTOR TRIP OR SAFETY INJECTION.

IF SI Actuation occurs during this procedure,

THEN GO TO EOP E-0, REACTOR TRIP OR SAFETY INJECTION.

3.0 **AFW SUPPLY SWITCHOVER CRITERION**

IF CST Level lowers to LESS THAN 10%,

THEN IMPLEMENT OP D-1:V, ALTERNATE AFW SUPPLIES.

Examination Outline Cross-Reference		**Level**	SRO
		Tier #	2
061 G2.4.4 – AFW: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.		**Group #**	1
		K/A #	061 G2.4.4
		Rating	4.7

Question 14

GIVEN:
- Unit 1 trips from 100% power
- The crew is performing EOP E-0, Reactor Trip or Safety Injection
- 4 kV Bus G is de-energized
- None of the MDAFW pumps can be started
- Narrow range steam generator levels are off scale low in all Steam Generators
- Safety Injection has not actuated

Which of the following action(s) should be taken by the Shift Foreman as the transition is made from EOP E-0?

A. Go to EOP E-0.1, Reactor Trip Response, the turbine driven AFW pump is supplying all four steam generators.

B. Go to EOP E-0.1, Reactor Trip Response, and direct the operator to locally open the TDAFW level control valves, LCV-106, 107, 108 and 109 to establish AFW flow from the TDAFW pump.

C. Go to EOP FR-H.1, Response to Loss of Secondary Heat Sink, there is a RED path on Secondary Heat Sink because the TDAFW Steam Supply valve, FCV-95 did not open due to the loss of Bus G.

D. Go to EOP FR-H.1, Response to Loss of Secondary Heat Sink, there is a RED path on Secondary Heat Sink because the TDAFW level control valves, LCV-106, 107, 108 and 109 are closed due to loss of Bus G.

Proposed Answer: A. Go to EOP E-0.1, Reactor Trip Response, the turbine driven AFW pump is supplying all four steam generators.

Explanation: NOTE: not a normal at power KA.

A. Correct. The TDAFW pump is running, it starts on 2/4 steam generators less than 15%.The TDAFW pump LCVs are powered from Bus G, however, are left full open and will remain open. Full flow from the pump will be supplying all four steam generators.

B. Incorrect. The valves will be fully open due to the loss of power and will have to locally throttled when narrow range level is greater than 16%. Plausible if its thought the valves must open and will not due to loss of power or that the valves fail closed on loss of power.

C. Incorrect. Unlike the MOVs for the TDAFW pump, FCV-95 is DC powered and will still have power to open and start the pump. Plausible if its thought that because of the loss of power, FCV-95 will not open and therefore, there is no AFW flow, resulting in a CSF heat sink RED pat.

D. Incorrect. The LCVs remain open despite the loss of power to them, therefore, there is no challenge to Heat Sink CSF. Plausible if its not known the valves are in the open position or if its thought they go closed on a loss of power – because they are MOVs – they will not move due to loss of power.

Technical References: LD-1, F-0

References to be provided to applicants during exam: None

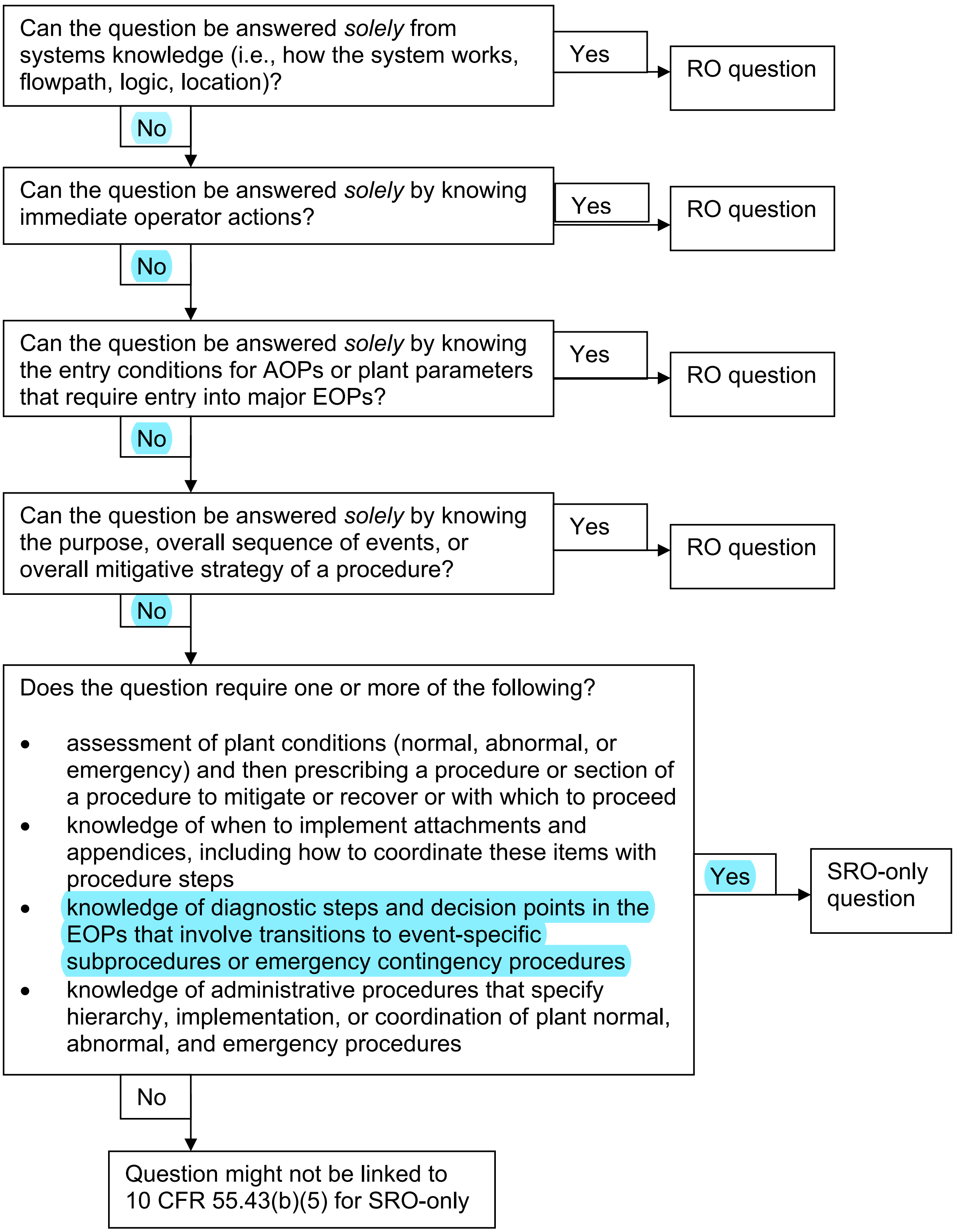
Learning Objective: 3552 Given initial conditions, assumptions, and symptoms, determine the correct Emergency Operating Procedure to be used to mitigate an operational event

Question Source:	Bank #89 L091C	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 03/2012	Y
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 3

Question 14

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Examination Outline Cross-Reference	Level	SRO
061 A2.07 Ability to (a) predict the impacts of the following malfunctions or operations on the AFW system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Air or MOV failure	**Tier #**	2
	Group #	1
	K/A #	061 A2.07
	Rating	3.5

Question SRO 14 (89) PARENT QUESTION

GIVEN:
- Unit 1 trips from full power
- The crew is performing E-0, Reactor Trip or Safety Injection
- 4 kV Bus G is de-energized
- None of MDAFW pumps can be started
- Narrow range steam generator levels are 12% in all Steam Generators
- Safety Injection has not actuated

Which of the following actions should be taken by the Shift Foreman as a transition is made from E-0?

A. Go to E-0.1, Reactor Trip Response, the turbine driven AFW pump is supplying all four steam generators.

B. Go to FR-H.1, Response to Loss of Secondary Heat Sink, there is a RED path on Secondary Heat Sink because the TDAFW level control valves, LCV-106, 107, 108 and 109 are closed due to loss of Bus G.

C. Go to E-0.1, Reactor Trip Response, and direct the operator to locally open the TDAFW level control valves, LCV-106, 107, 108 and 109 to establish AFW flow from the TDAFW pump.

D. Go to FR-H.1, Response to Loss of Secondary Heat Sink, there is a RED path on Secondary Heat Sink because the TDAFW Steam Supply valve, FCV-95 did not open due to the loss of Bus G.

Proposed Answer: A. Go to E-0.1, Reactor Trip Response, the turbine driven AFW pump is supplying all four steam generators.

Explanation:

A. Correct. The TDAFW pump is running, it starts on 2/4 steam generators less than 15%.The TDAFW pump LCVs are powered from Bus G, however, are left full open and will remain open. Full flow from the pump will be supplying all four steam generators.
B. Incorrect. The LCVs remain open despite the loss of power to them, therefore, there is no challenge to Heat Sink CSF.
C. Incorrect. The valves will be fully open due to the loss of power and will have to locally throttled when heat sink is greater than 16%.
D. Incorrect. FCV-95 is DC powered.

Technical References: LD-1, F-0

References to be provided to applicants during exam: None

Learning Objective: 8405 State the power supplies to Auxiliary Feed Water System components.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.	

Editorial – replaced "from low power" to "from full power". This ensures that the low levels in the steam generator are an expected response.

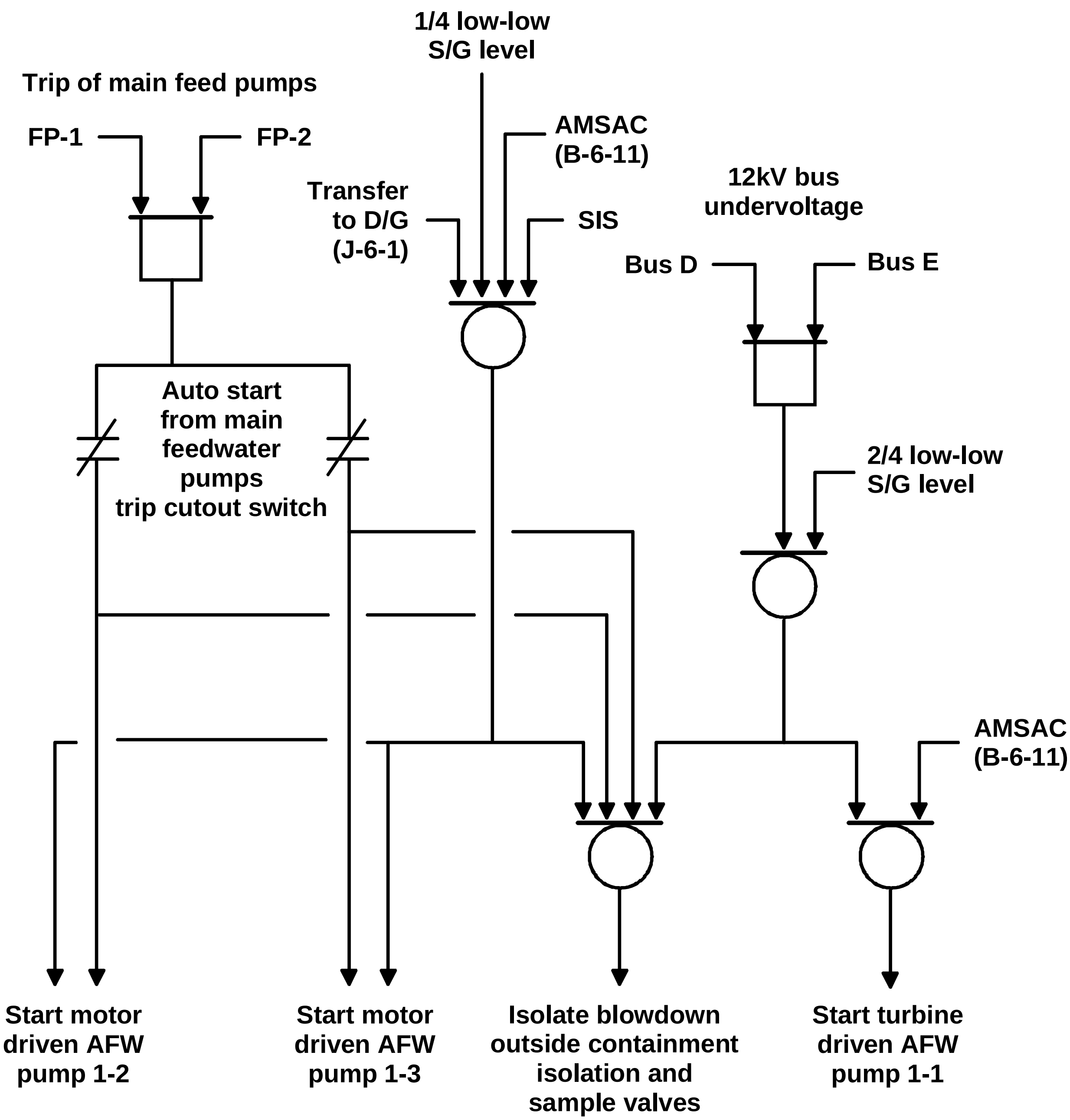
Auxiliary Feed Pump Start Signals

Motor Driven Pumps

1. AMSAC (B-6-11)
2. <15% level (2/3) in 1/4 S/G's with time delay ***
3. Transfer to diesel generator (J-6-1)
4. Both MFP's tripped (can be cutout on VB3)
5. Safety Injection signal

Turbine Driven Pump

1. AMSAC (B-6-11)
2. <15% level (2/3) in 2/4 S/G's with time delay ***
3. 12 kV bus undervoltage (1/2 sensors on 2/2 buses)



*** See Note at Bottom of Page B-6-4b

Ref: 495855

TDAFW Pump Steam Supply Valve FCV-95

Objective 5

State the purpose of Auxiliary Feed Water System components.

- TDAFW Pump Steam Supply Valve FCV-95

Study Topics

The purpose of FCV-95 is to provide isolation of the steam supply to the TDAFW pump and maintain it in a standby condition.

Objective 6

Identify the location of components associated with the Auxiliary Feed Water System.

- TDAFW Pump Steam Supply Valve FCV-95

Study Topics

FCV-95 is located in the 115' penetration area.

Objective 7

State the power supplies to Auxiliary Feed Water System components.

- TDAFW Pump Steam Supply Valve FCV-95

Study Topics

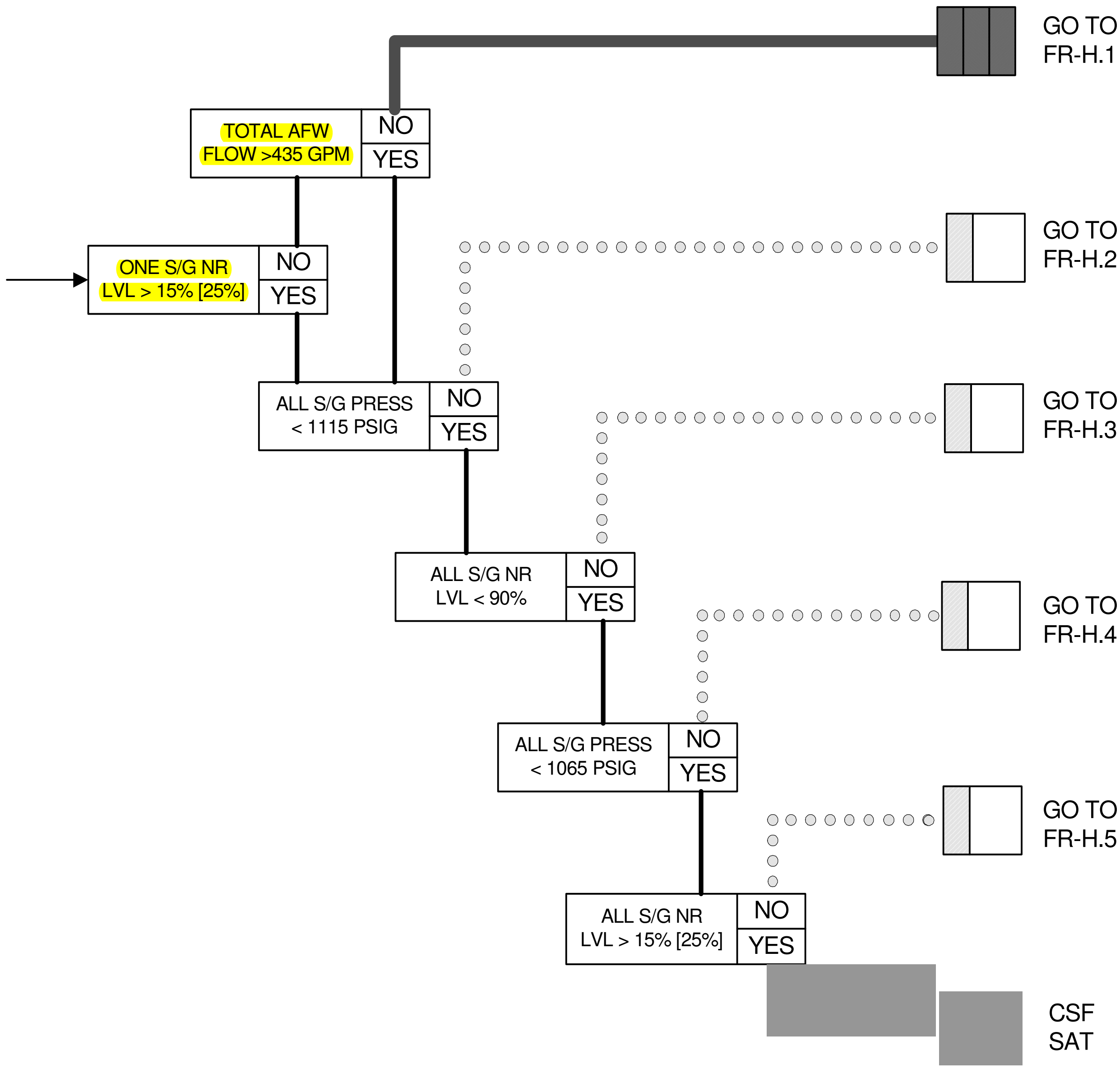
FCV-95 is powered from DC Panel 12.

Continued on next page

TDAFW Pump LCVs LCV-106, LCV-107, LCV-108, and LCV-109

Objective 5	State the purpose of Auxiliary Feed Water System components. • TDAFW Pump LCVs LCV-106, LCV-107, LCV-108, and LCV-109
Study Topics	**The purpose of the TDAFW pump LCVs is to control AFW flow to all four SGs and to provide isolation capability of a faulted SG.**
Objective 6	Identify the location of components associated with the Auxiliary Feed Water System. • TDAFW Pump LCVs LCV-106, LCV-107, LCV-108, and LCV-109
Study Topics	**LCV-106 and LCV-107 are located in 115' pipe rack area. LCV-108 and LCV-109 are located in the 115' penetration area.**
Objective 7	State the power supplies to Auxiliary Feed Water System components. • TDAFW Pump LCVs LCV-106, LCV-107, LCV-108, and LCV-109
Study Topics	**The power supply to all four LCVs is 480 V bus G.**
Objective 8	Describe Auxiliary Feed Water components. • TDAFW Pump LCVs LCV-106, LCV-107, LCV-108, and LCV-109
Study Topics	**LCV-106 through LCV-109 are motor operated Limitorque valves. The valves are manually controlled and are normally left full open.**

F-0.3 HEAT SINK



Examination Outline Cross-Reference		**Level**	SRO
		Tier #	2
073 A2.01 Ability to (a) predict the impacts of the following on the Process Radiation Monitoring System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: PRM components failures		**Group #**	1
		K/A #	073 A2.01
		Rating	3.1

Question 15

GIVEN:

- Unit 1 is at 100% power
- A discharge of Process Waste Receiver 0-1 is in progress per OP G-1:II, Liquid Radwaste System - Discharge of Liquid Radwaste

RE-18, Liquid Rad Waste Discharge Radiation Monitor, loses power during the discharge. RE-18 is subsequently declared inoperable.

1) The discharge will:

2) The discharge while RE-18 is inoperable may recommence once the lineup is verified and:

A. 1) continue until terminated by the operator.
   2) approval by the Station Director is obtained to enter ECG 0.3

B. 1) continue until terminated by the operator.
   2) two independent samples are taken and analyzed by two technically qualified technicians.

C. 1) be automatically terminated when RE-18 loses power.
   2) approval by the Station Director is obtained to enter ECG 0.3

D. 1) be automatically terminated when RE-18 loses power.
   2) two independent samples are taken and analyzed by two technically qualified technicians.

Proposed Answer: D. be automatically terminated when RE-18 loses power..
2) two independent samples are taken and analyzed by two technically qualified technicians

Explanation:

A. Incorrect. First part plausible because the indication fails low – could be that this would correlate to a low reading and the discharge would not stop. However, loss of power causes the relay to actuate and terminate the discharge. Second part is incorrect but plausible if its not known that there is an action in ECG 39.3 if the rad monitor is OOS. ECG 0.3 is the action that can be entered if there is no action provided in the ECG.
B. Incorrect. First is incorrect, the release will be automatically terminated due to the loss of power. Plausible because the second part is correct.
C. Incorrect. First part is correct, the release will terminate. Second part incorrect, a discharge may continue if two samples are drawn and independently verified per ECG 39.3. Plausible as ECG 0.3 is used to address situations where there is not an applicable action in

the ECG (similar to LCO 3.0.3). This would be the action if ECG 39.3 did not cover the rad monitor inoperability.

D. Correct. The release will terminate automatically when power is lost to the rad monitor. If RE-18 is inoperable, releases may occur if the actions stated occur. ECG 39.3 action A are taken.

Technical References: ECG-39.3, ECG 0.0, LG-4A

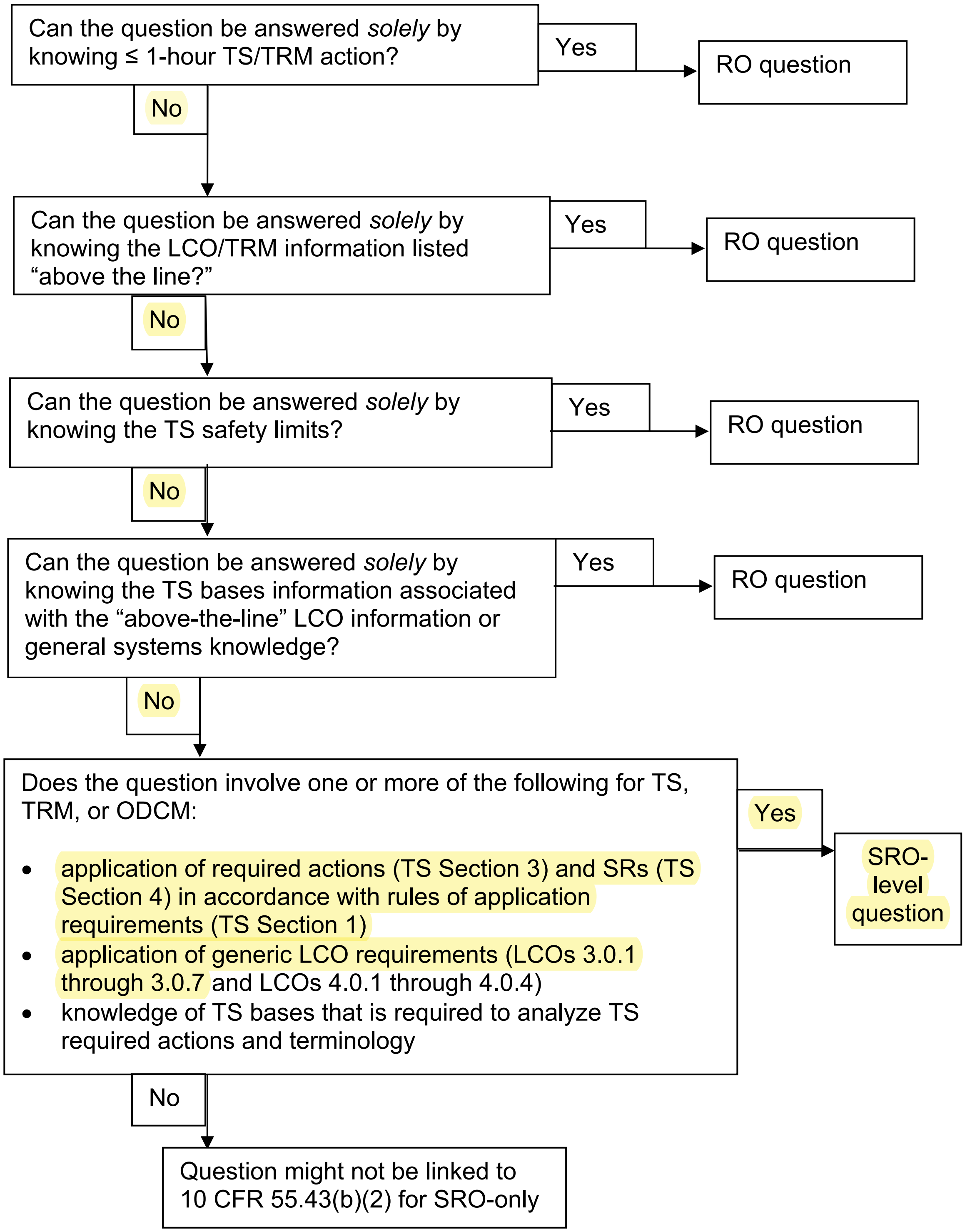
References to be provided to applicants during exam: None

Learning Objective: Apply the requirements of System 39 ECGs,

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #93 L121	X
	New	
	Past NRC Exam DCPP 08/14	Y
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.1	

Difficulty: 3

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



c. *Facility Licensee Procedures Required to Obtain Authority for Design and Operating Changes in the Facility [10 CFR 55.43(b)(3)]*

Some examples of SRO-only examination items for this topic include the following:

Examination Outline Cross-Reference	Level	SRO
068 A2.04 - Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of automatic isolation	**Tier #**	2
	Group #	2
	K/A #	068 A2.04
	Rating	3.3

Question 93 PARENT QUESTION

GIVEN:

- A discharge of Process Waste Receiver 0-1 is in progress per OP G-1:II, Liquid Radwaste System - Discharge of Liquid Radwaste
- A High Radiation alarm is received for RE-18, Liquid Radwaste Radiation Detector
- The release fails to terminate automatically due to the high radiation signal but is terminated by the operator at the Aux Panel in less than a minute
- Due to the failure to automatically terminate the release, RE-18 is declared inoperable

Which of the following:
1) Identifies the Federal Regulatory impact, if any, of this condition?
2) If the discharge may occur while RE-18 is inoperable?

A. 1) The continued release may have exceeded the Liquid Effluent Limits of the Offsite Dose Calculation Procedure.
2) No releases may occur until RE-18 is OPERABLE.

B. 1) The continued release may have exceeded the Liquid Effluent Limits of the Offsite Dose Calculation Procedure.
2) The discharge may occur provided two independent samples are taken and analyzed by two technically qualified technicians.

C. 1) No regulatory impact, the discharge permit is a state, not a federal, regulation.
2) No releases may occur until RE-18 is OPERABLE.

D. 1) No regulatory impact, the discharge permit is a state, not a federal, regulation.
2) The discharge may occur provided two independent samples are taken and analyzed by two technically qualified technicians.

Proposed Answer: B. The continued release may have exceeded the Liquid Effluent Limits of the Offsite Dose Calculation Procedure.
2) The discharge may occur provided two independent samples are taken and analyzed by two technically qualified technicians.

Explanation:

A. Incorrect. Provisions allow for releases with inoperable radiation monitors.

B. Correct. The ODCM is a part of the plant license. If RE-18 is inoperable, releases may occur if the actions stated occur. ECG 39.3 bases states: The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the Offsite Dose Calculation procedure (CAP A-8) to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20.

C. Incorrect. The ODCM is part of the plant license.

D. Incorrect. The ODCM is part of the plant license.

Technical References: ECG-39.3, Tech Spec 5.5. LM-8

References to be provided to applicants during exam: None

Learning Objective: 66068 - Apply the requirements of System 39 ECGs.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.1	

RM-17 A/B, RM-18, RM-19, RM-22 and RM-23 Process Monitors, Continued

Objective 10 Analyze automatic features and interlocks associated with the Radiation Monitoring System.

- RM-17 A/B, RM-18, RM-19, RM-22 and RM-23 Process Monitors

Study Topics **Process Monitor automatic actions**

A high alarm (or loss of power) on RM-17 A or B will cause:

- **The CCW System surge tank vent valve, RCV-16 to close.**
  - **This prevents a gaseous radioactivity release through the CCW system.**
  - **loss of power - meter will be at minimum scale**

A high alarm (or loss of power) on RM-18 will cause:

- **The LRW System discharge isolation valve, RCV-18 to close.**
  - **The LRW System will recirculate to the Equipment Drain Receiver via FCV-477.**
  - **loss of power - meter will be at minimum scale**

- **A high alarm (or loss of power) on RM-22 will cause the gas decay tank discharge to the plant vent to isolate (RCV-17) and FCV-410,404,405, and 406 will close and are blocked from opening.**
  - **loss of power - meter will be at minimum scale**

A high alarm (or loss of power) on RM-19 or RM-23 will cause:

- **The SGBD isolation valves, outside containment, to close.**
- **The SGBD sample isolation valves to close.**
- **SGBD from the Blow Down Tank will be realigned from the discharge tunnel to the Equipment Drain Receiver.**
- **loss of power - meter will be at minimum scale**

Note: The defeat toggle switch on VB-3 will allow the sample isolation valves to be reopened for sampling to determine a faulted SG.

Continued on next page

0.0 APPLICABILITY

EQUIPMENT CONTROL GUIDELINES

ECG 0.1	Equipment Control Guidelines (ECG) shall be met during the MODES or other specified conditions in the Applicability, except as provided in ECG 0.2.
ECG 0.2	Upon discovery of a failure to meet an ECG, the associated ACTIONS shall be met. If the ECG is restored prior to expiration of the specified Completion Time(s), completion of the Required Action is not required, unless otherwise stated.
ECG 0.3	When an ECG is not met, and the associated ACTIONS are not met or an associated ACTION is not provided, initiate a Notification, as appropriate, and obtain station director approval per OP1.DC16. Station director approval should be obtained prior to entering ECG 0.3 and shall be obtained within 24 hours of entering ECG 0.3. Where corrective measures are completed that permit operation in accordance with the ECGs, completion of the actions required by ECG 0.3 are not required.
ECG 0.4	When an ECG is not met, entry into a MODE or other specified condition in the Applicability shall only be made: a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time; or b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this ECG are stated in the individual ECG, or c. When an allowance is stated in the individual value, parameter, or other ECG. This ECG shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.
ECG 0.5	Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to ECG 0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

39.0 INSTRUMENTATION

39.3 Radioactive Liquid Effluent Monitoring Instrumentation

ECG 39.3 The following radiation monitors with associated flow channels shall be operable with their alarm/trip setpoints set to ensure that the limits of the Radiological Monitoring and Control Program (IDAP CY2.ID1) are not exceeded:

RM-3	RM-18	RM-23
FR-251	FIT-243	FR-53

The alarm/trip setpoints of these channels shall be determined in accordance with the methodology and parameters in the offsite dose calculation procedure (CAP A-8).

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
----------------NOTE----------------- RM-18 is common to both units --- A. Liquid Radwaste Effluent Line (RM-18) Inoperable	-----------------NOTE (A)-------------- Effluent releases via this pathway may continue for up to 14 days provided that prior to initiating a release: a. At least two independent samples are analyzed in accordance with IDAP CY2.ID1 and b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge line valvings. Otherwise, comply with Action A.1 ---	

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.1 Suspend release of radioactive effluents via this pathway <u>AND</u> A.2.1 Restore RM-18 to operable status (see note A) <u>OR</u> A.2.2 Explain in Effluent Release Report pursuant to TS 6.9.1.6 (ITS 5.6.3) why this inoperability was not corrected within the time specified.	Immediately 14 days Next submittal of the Effluent Release Report
B. Steam Generator Blowdown Tank (RM-23) inoperable	-----------------NOTE (B)-------------- Effluent releases via this pathway may continue for up to 30 days provided grab samples are analyzed for radioactivity (beta or gamma) at a lower limit of detection of no more than $5x10^{-7}$ microcuries/ml: a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microcuries/gm dose equivalent I-131, <u>OR</u> b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcuries/gm dose equivalent I-131. Otherwise, comply with Action B.1. ---	

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	B.1 Suspend release of radioactive effluents via this pathway AND B.2.1 Restore RM-23 to operable status (see note B) OR B.2.2 Explain in Effluent Release Report pursuant to TS 6.9.1.6 (ITS 5.6.3) why this inoperability was not corrected within the time specified.	Immediately 30 days Next submittal of the Effluent Release Report.
C. One or more of the following flow channels are inoperable: Liquid Radwaste Effluent Line (FIT-243) Steam Generator Effluent Lines (FR-53) Oily Water Separator Effluent Line (FR-251) ---------------NOTE------------------- FIT-243 and FR-251 are common to both units ---	--------------------NOTE (C)--------------- Effluent releases via the pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves may be used to estimate flow. Otherwise, comply with Action C.1. --- C.1 Suspend release of radioactive effluents via the pathway. AND C.2.1 Restore the inoperable flow rate measurement devices to operable status (see note C) OR C.2.2 Explain in Effluent Release Report pursuant to TS 6.9.1.6 (ITS 5.6.3) why the inoperability was not corrected within the time specified.	 Immediately 30 days Next submittal of the Effluent Release Report

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Oily Water Separator Effluent Line (RM-3) inoperable	----------------NOTE (D)------------- Effluent releases via this pathway may continue for up to 30 days provided that: a. At least once per 12 hours, grab samples are collected and analyzed for radioactivity (beta or gamma) at a lower limit of detection of no more than $5x10^{-7}$ microcuries/ml, <u>OR</u> b. The oily water separator effluent is transferred to the Liquid Radwaste Treatment System. Otherwise, comply with Action D.1. --	
	D.1 Suspend release of radioactive effluents.	Immediately
	<u>AND</u>	
	D.2.1 Restore RM-3 to operable status (see note D)	30 days
	<u>OR</u>	
	D.2.2 Explain in Effluent Release Report pursuant to TS 6.9.1.6 (ITS 5.6.3) why this inoperability was not corrected within the time specified.	Next submittal of the Effluent Release Report

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 39.3.1	----------NOTE---------- For FIT-243, FR-53 and FR-251 a CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK for FR-251 shall be made once per calendar day*, and for FIT-243 and FR-53 shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made. -------------------- Perform CHANNEL CHECK on the following: RM-3 RM-18 RM-23 FR-251 FIT-243 FR-53	24 hours for FIT-243 and FR-53. Once per calendar day* for FR-251.
SR 39.3.2	Perform a SOURCE CHECK on the following: RM-18	Prior to each release.
SR 39.3.3	Perform a SOURCE CHECK on the following: RM-3 RM-23	31 days
SR 39.3.4	----------NOTE---------- For RM-18, RM-23, and RM-3 the initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Institute of Standards and Technology (NIST) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used. --------------------	18 months
	Perform CHANNEL CALIBRATION on the following: RM-3 RM-18 RM-23 FR-251 FIT-243 FR-53	

(continued)

* The frequency "once per calendar day" could result in two successive channel checks nearly 48 hours apart over a two day period. This frequency is different from and should not be confused with the frequency notation "D" (at least once per 24 hours) defined in Technical Specifications.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 39.3.5	------------------------------NOTE------------------------------------ For RM-18 and RM-23 the CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists: a. Instrument indicates measured levels above the Alarm/Trip Setpoint (isolation and alarm), or b. Relay control circuit failure (isolation only), or c. Instrument indicates a downscale failure (alarm only), or d. Instrument controls not set in operate mode (alarm only). For RM-3 the CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists: a. Instrument indicates measured levels above the Alarm Setpoint, or b. Circuit failure, or c. Instrument indicates a downscale failure, or d. Instrument controls not set in operate mode. --- Perform CHANNEL FUNCTIONAL TEST on the following: RM-3 RM-18 RM-23 FR-251 FIT-243 FR-53	92 days
SR 39.3.6	Verify one saltwater pump is operating and providing dilution to the discharge structure.	4 hours

BASES

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the Offsite Dose Calculation Procedure (CAP A-8) to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria (GDC): GDC 17, 1967; GDC 18, 1967; and GDC 70, 1967.

All portions of each flow recorder or channel (FR-251, FIT-243, and FR-53 loops) are required for OPERABILITY, including the associated totalizer, digital displays, and pens of the recorders, as applicable.

The daily FR-251 CHANNEL CHECK required by SR 39.3.1 should be performed, if possible, by observing an actual OWS automatic discharge. If no automatic discharge occurs during a given day, then late in the day a manual discharge could be initiated to allow observation of FR-251. FR-251 CHANNEL CHECKs on two successive days could be up to 48 hours apart. An FR-251 CHANNEL CHECK is not required for each OWS discharge.

REFERENCES

1. IDAP CY2.ID1, "Radioactive Effluent Controls Program."
2. CAP A-8, "DCPP Offsite Dose Calculation Procedure" (ODCP).
3. DN 50032741-FR-53 Inoperable and conditional surveillance.

Examination Outline Cross-Reference		**Level**	SRO
		Tier #	2
002 A2.04 Ability to (a) predict the impacts of the following on the Reactor Coolant System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of Heat Sinks		**Group #**	2
		K/A #	002 A2.04
		Rating	4.5

Question 16

According to the PTLR for Diablo Canyon, prior to starting the first RCP in one loop:

A. Pressurizer level must be greater than 56%.

B. Steam generator narrow level must be at least 15%.

C. RCS temperature must be at least 50°F higher than steam generator temperature.

D. Steam generator temperature must be less than 50°F higher than RCS temperature.

Proposed Answer: D. Steam generator temperature must be less than 50°F higher than RCS temperature.

Explanation:

KA is met because if the steam generators are at a greater temperature than the RCS, they are a heat SOURCE and not acting as a heat SINK. The PLTR outlines the case for LTOP Heat injection case that it is caused by starting a RCP with a steam generator(s) 50°F above RCS temperature, causing a volumetric change in RCS. This can be mitigated if pressurizer level is below 50% to accommodate the expansion. This question asks for basis from the PTLR for a condition of steam generator temperature above RCS temperature.

SRO Justification (in lieu of worksheet) from ES-4.2 - 3. Examination authors should use the 10 CFR 55.43(b) topic-based guidance and examples (a–g below) when developing SRO-level questions.

a. Conditions and Limitations in the Facility License [10 CFR 55.43(b)(1)]

Additionally, Tier 2 questions should focus on normal activities when possible.

A. Incorrect. RCP starting requirements in EOPs usually require a higher pressurizer level to accommodate for outsurge. The concern here is heatup or insurge and the requirement that would apply is level less than 50%.
B. Incorrect. In many EOPs, like FR-C.1, to start a RCP in a loop, there must be a heat sink (level greater than 15%). 15% defines what is necessary for the steam generator to be a heat sink.
C. Incorrect. While it could be thought that the steam generators would be a heat sink, the PTLR sets limits for starting RCPs when the RCS is colder than the steam generators – when they are a heat sink and would cause a heatup.
D. Correct. The heat injection cases establish that there are no LTOP administrative RCS temperature restrictions for starting an RCP when the measured SG temperature does not exceed the RCS by more than 50°F. A bounding heat injection case was also evaluated to establish that if the pressurizer level indicates less than or equal to 50%, there are no RCS/SG temperature restrictions for starting an RCP, since even the maximum credible

RCS/SG temperature differential will not challenge the Appendix G P/T limit in the LTOP range.

Technical References: PTLR

References to be provided to applicants during exam: None

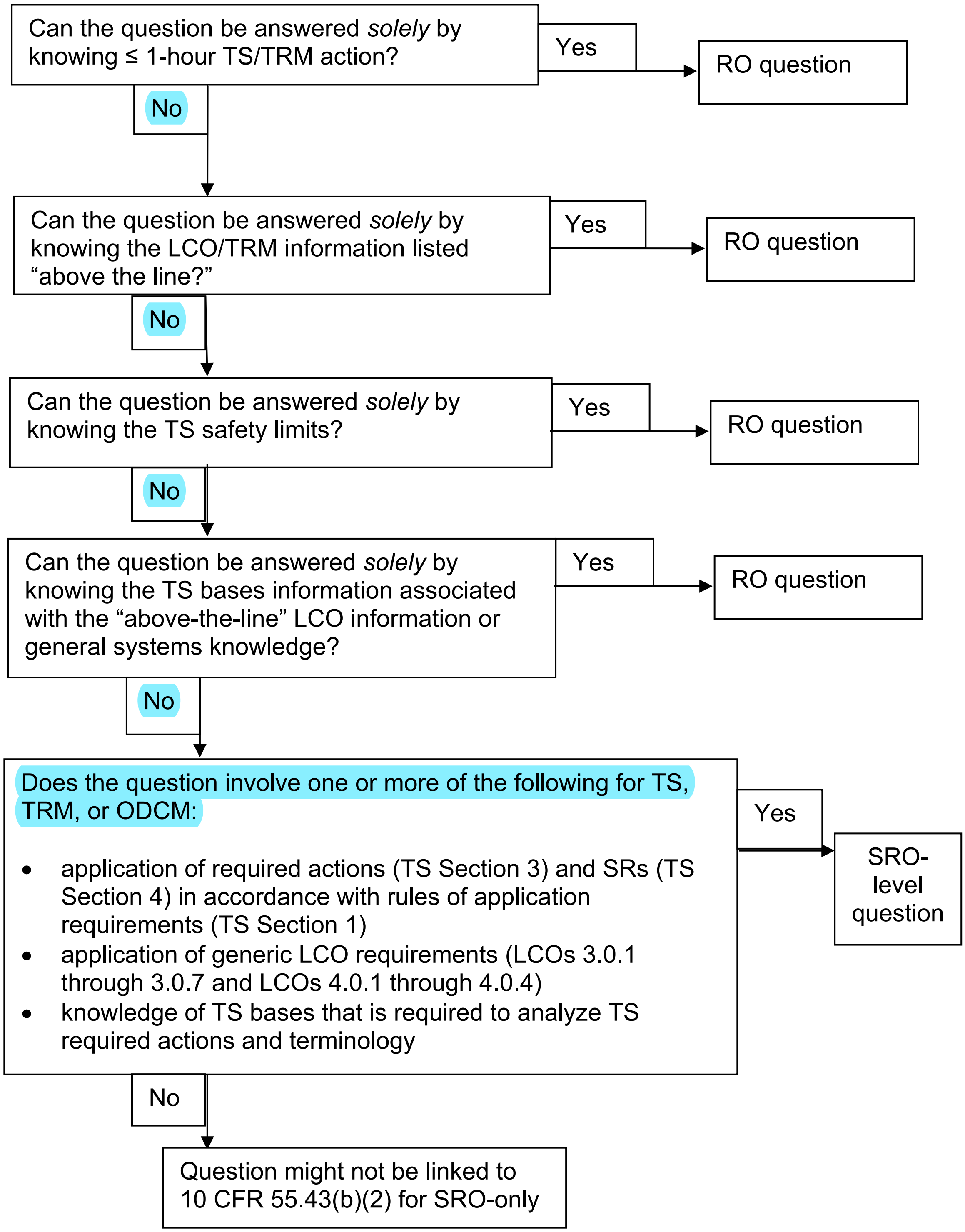
Learning Objective: 65720 - Describe the operating limits found in the Pressure Temperature (P/T) Limits Report (PTLR) and the related Technical Specifications (TS).

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	N
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.1	

Difficulty: 2.5

Question 16

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



c. *Facility Licensee Procedures Required to Obtain Authority for Design and Operating Changes in the Facility [10 CFR 55.43(b)(3)]*

Some examples of SRO-only examination items for this topic include the following:

2.2.4 LTOP Heat Injection Case:

The heat injection cases are based on starting an RCP in one loop with a maximum allowable measured temperature difference of 50 °F between the RCS and the Steam Generators (SGs). The heat injection cases are evaluated at various RCS temperature conditions which bound the potential volumetric expansion effects of water on the RCS overshoot within the LTOP range. The heat injection RCS overshoot cases were determined to remain below the Appendix G P/T curve and are conservatively bounded by the mass injection overshoot results throughout the LTOP temperature range. The heat injection cases establish that there are no LTOP administrative RCS temperature restrictions for starting an RCP when the measured SG temperature does not exceed the RCS by more than 50 °F. A bounding heat injection case was also evaluated to establish that if the pressurizer level indicates less than or equal to 50%, there are no RCS/SG temperature restrictions for starting an RCP, since even the maximum credible RCS/SG temperature differential will not challenge the Appendix G P/T limit in the LTOP range.

2.2.5 RCS Pressure Undershoot:

Once an LTOP PORV has opened to mitigate the pressure transient due to a mass injection or heat injection case, the RCS pressure continues decreasing even after the close setpoint has been reached and until the PORV has fully closed. The limiting RCS undershoot case is based on the maximum RCS pressure relief capacity associated with both LTOP PORVs opening and closing simultaneously during the least severe mass injection and heat injection overshoot case, respectively. The RCS undershoot evaluation is based on maintaining the RCS pressure above the minimum value which is considered acceptable for the number one RCP seal operating conditions. The PORV lift setpoint in Table 2.2-1 was evaluated to adequately limit the RCS undershoot to an acceptable value for the applicable mass injection and heat injection cases within the LTOP range.

Where there is insufficient range between the upper and lower pressure limits to select a PORV setpoint to provide protection against violation of both limits, setpoint selection to provide protection against the upper pressure limit violation shall take precedence.

Examination Outline Cross-Reference	**Level** SRO
034 G2.3.14 FHES: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities, such as analysis and interpretation or radiation and activity readings as they pertain to administrative, normal, abnormal, and emergency procedures or to analysis and interpretation of coolant activity, including comparison to emergency plan or regulatory limits. (SRO only)	**Tier #** 2 **Group #** 2 **K/A #** 034 G2.3.14 **Rating** 3.8

Question 17

A plant shutdown for refueling is being performed.

1. In accordance with ECG 42.1 Refueling Operations - Decay Time, the reactor shall be subcritical for at least ______ prior to moving irradiated fuel in the reactor vessel.
2. Per ECG 42.1 bases, this time delay requirement ensures that if the postulated fuel handling accident occurs, ____ are maintained within allowable limits.

A. 1. 72 hours
   2. offsite doses

B. 1. 72 hours
   2. radiation exposure to personnel in containment

C. 1. 100 hours
   2. offsite doses

D. 1. 100 hours
   2. radiation exposure to personnel in containment

Proposed Answer: C. 1. 100 hours 2. offsite doses

Explanation:
The KA is met as the bases for the minimum decay time is a regulatory limit and the fuel handling accident is a radiation hazard that could occur during fuel movement.

A. Incorrect. 72 hours is in the bases for 100 hour requirement. The accident is assumed to occur within 72 hours. Waiting 100 hours ensures the results of the accident are acceptable and maintain offsite doses within allowable limits. Second part is correct.
B. Incorrect. Both parts are incorrect. Plausible because ALARA is a goal for the station, protecting the public is what the analysis ensures. Also, in the bases for 3.7.10, Control Room Ventilation System, purpose of the system is "The CRVS provides airborne radiological protection **for the CRE occupants**, as demonstrated by the CRE occupant dose analyses for the most limiting **design basis accident, fission product release** presented in the UFSAR, Chapter 15 (Ref. 2)"., so thinking the same could apply for personnel in containment when an event occurs is plausible.
C. Correct. The 100 hour requirement ensures that if the fuel handling accident occurs, (assumed in the first 72 hours) the accident results will be acceptable.
D. Incorrect. First part is correct.

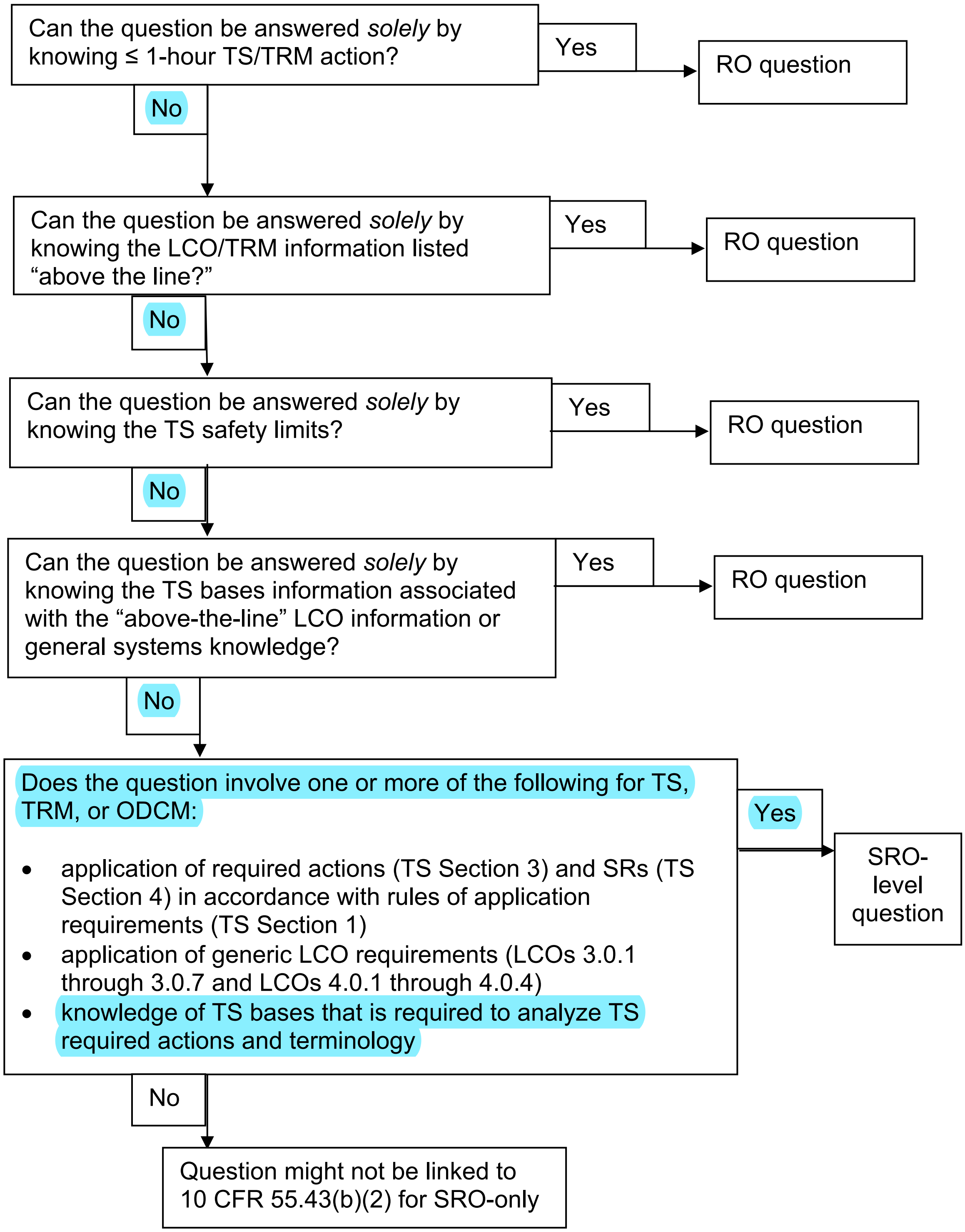
Technical References: ECG 42.1, LCO 3.7.10 bases
References to be provided to applicants during exam: None
Learning Objective: 66070 Apply the requirements of System 42 ECGs.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	N
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.2	

Difficulty: 2.0

Question 17

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



c. *Facility Licensee Procedures Required to Obtain Authority for Design and Operating Changes in the Facility [10 CFR 55.43(b)(3)]*

Some examples of SRO-only examination items for this topic include the following:

42.0 FUEL HANDLING SYSTEM

42.1 Refueling Operations - Decay Time

ECG 42.1 The reactor shall be subcritical for at least 100 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor vessel.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor subcritical for <100 hours.	A.1 Suspend all operations involving movement of irradiated fuel in the reactor vessel.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 42.1.1	Verify reactor subcritical ≥ 100 hours.	Prior to movement of irradiated fuel in the reactor vessel.

BASES

BACKGROUND	The minimum time requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of short lived fission products. In addition, this minimum decay time is input data used in calculating the decay heat load in the spent fuel pool (SFP) as a basis for determining the requirements of the SFP cooling system.
APPLICABLE SAFETY ANALYSES	Prior to moving irradiated fuel assemblies, the reactor is assumed to have operated at 3580 MWt with current licensed values of fuel enrichment and fuel burnup (References 12 and 13). A fuel handling accident is assumed to occur 72 hours after shutdown. This is conservative since this decay time is less than the minimum time required by this ECG. DCPP procedures prohibit movement of recently irradiated fuel which is defined as fuel that has occupied part of a critical reactor core within the previous 100 hours. The assumed gap activity inventory in a single fuel assembly at 72 hours post reactor shutdown is described in References 1 and 3. This inventory conservatively bounds that associated with the 100-hour time period required by this ECG. The key assumptions in the fuel handling accident are described in Reference 3. The fuel handling accident inside containment is described in Reference 3. With a minimum decay time of at least 72 hours and minimum refueling cavity water level of 23 ft prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained well within allowable limits (References 5 and 6). The thermal-hydraulic design basis of the SFP cooling system is described in the UFSAR. Though unlikely, in the event of a loss of forced cooling due to failure of the SFP heat exchanger, surface cooling would maintain the water temperature at or below the boiling point (Reference 10).

(continued)

BASES (continued)

LCO	A decay time of at least 100 hours is required to ensure that the radiological consequences of a postulated fuel handling accident are within acceptable limits (References 6 and 14). In addition, the 100-hour decay time ensures that the SFP cooling system can effectively maintain bulk pool temperatures below 175°F, thus providing a pool boiling margin for a postulated loss of pool cooling (References 7, 8, and 11). The time to boil of 2.5 hours (full core off-load) is based on the 100-hour decay time (References 8 and 9).
APPLICABILITY	ECG 42.1 is applicable in MODE 6 during handling of irradiated fuel assemblies in the reactor vessel. The ECG does not allow for the fuel handling system to be used if the reactor has been subcritical for less than 100 hours. This allows time for short lived fission products in the reactor core to decay. This results in reduced fuel handling personnel exposure and reduces the decay heat load demands on the spent fuel pool cooling system. The radiological dose consequences are based on a 72 hour decay time (References 6 and 14).
ACTIONS	A.1 With decay time less than 100 hours, all operations involving movement of irradiated fuel assemblies within the reactor vessel shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of fuel movement shall not preclude completion of movement of a component to a safe position.
SURVEILLANCE REQUIREMENTS	SR 42.1.1 Verification of subcriticality for at least 100 hours prior to movement of irradiated fuel in the reactor vessel ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. This verification also ensures that the decay heat load assumptions used in the design basis SFP cooling analyses are met. The verification shall be done by verifying the date and time of subcriticality.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 15.5.22, "Radiological Consequences of a Fuel Handling Accident"
2. Regulatory Guide 1.183, July 2000
3. UFSAR, Section 15.4.5, "Fuel Handling Accident"
4. NUREG-0800, Section 15.7.4
5. 10 CFR 50.67
6. 14078104-C-M-00011, Rev. 1, “Site Boundary and Control Room Doses Following a Fuel Handling Accident in Containment or Fuel Handling Building using Alternative Source Terms” [SAP Calc Series No. 6024616-11]
7. NRC SER, October 16, 1974
8. "Report on Reracking of Spent Fuel Pools for Diablo Canyon Units 1 and 2." (Enclosure to PG&E Letter DCL-85-306, September 19, 1985)
9. License Amendments 6/8, dated May 30, 1986
10. UFSAR, Section 9.1.3.3.6, "General Design Criterion 67, 1967 – Fuel and Waste Storage Decay Heat"
11. License Amendments 183/185, dated November 21, 2005
12. UFSAR , Section 15.5.3, “Activity Inventories in the Plant Prior to Accidents”
13. 14078104-C-M-00007, Rev. 0, “Composite Equilibrium Reactor Core Isotopic Inventory Assuming an Initial U-235 Enrichment of 4.2% to 5%” [SAP Calc Series No. 6024616-7]
14. License Amendments 230/232, dated April 27, 2017

BASES

BACKGROUND (continued)	Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CRVS is designed in accordance with Seismic Category I requirements. The CRVS is designed to maintain a habitable environment in the CRE for the duration of the most severe Design Basis Accident (DBA) without exceeding a 5 rem TEDE dose.
APPLICABLE SAFETY ANALYSES	The CRVS components are arranged in redundant, PG&E Design Class I ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access. The CRVS provides airborne radiological protection for the CRE occupants, as demonstrated by the CRE occupant dose analyses for the most limiting design basis accident, fission product release presented in the UFSAR, Chapter 15 (Ref. 2). There are no offsite or onsite hazardous chemicals that would pose a credible threat to control room habitability. Consequently, engineered controls for the control room are not required to ensure habitability against a hazardous chemical threat. The amount of CRE unfiltered inleakage is not incorporated into PG&E's hazardous chemical assessment. The evaluation of a smoke challenge demonstrated that smoke will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panels (Ref. 1). The assessment verified that a fire or smoke event anywhere within the plant would not simultaneously render the Hot Shutdown Panel (HSP) and the CRE uninhabitable, nor would it prevent access from the CRE to the HSP in the event remote shutdown is required. No CRVS automatic actuation is required for hazardous chemical releases or smoke and no Surveillance Requirements are required to verify operability in cases of hazardous chemicals or smoke.

(continued)

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	2
045 A2.16 Ability to (a) predict the impacts of the following on the Main Turbine Generator System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Turbine blade failure	**Group #**	2
	K/A #	045 A2.16
	Rating	3.3

Question 18

Unit 1 is at 75% power, ramping to 100%, in accordance with OP L-4, Normal Operation at Power.

PK12-17, Turbine Supervision alarms, input 1109, Turbine Supervisor Instru Drawer, due to high vibration. The operator reports the following vibration readings:
- Vibration on bearing 1 has taken a step change from 1 to 12 mils
- Vibration on bearing 2 is 8 mils

1. In accordance with AR PK12-17, this is an indication of:
2. What action should be taken by the Shift Foreman?

A. 1. loss of oil to the bearings.
   2. Direct the operator to trip the reactor and go to EOP E-0, Reactor Trip or Safety Injection.

B. 1. loss of oil to the bearings.
   2. Direct the operator to trip the turbine and break vacuum in accordance with OP AP-29, Main Turbine Malfunction.

C. 1. a high pressure turbine blade being thrown.
   2. Direct the operator to trip the reactor and go to EOP E-0, Reactor Trip or Safety Injection.

D. 1. a high pressure turbine blade being thrown.
   2. Direct the operator to trip the turbine and break vacuum in accordance with OP AP-29, Main Turbine Malfunction.

Proposed Answer: C. 1. a high pressure turbine blade being thrown.
2. Direct the operator to trip the reactor and go to EOP E-0, Reactor Trip or Safety Injection.

Explanation:
A. Incorrect. Loss of oil could cause high vibration but the ARPK states a step of 10 mils on bearing 1 or 2 is an indication of a thrown blade. Second part, action is correct.
B. Incorrect. Both are incorrect. Action is plausible as in OP AP-29, in multiple sections, there is a caution that states "Breaking vacuum after the Turbine trip will reduce turbine coastdown time. However, this should be done only as a last resort, since breaking vacuum with Turbine speed greater than 200 rpm could cause serious Turbine damage." A loss of bearing oil or thrown blade could be viewed as a time vacuum should be broken above 200

rpm. However, entry into OP AP-29 is not appropriate - with power above P-9, for any instance a turbine trip is required, the reactor is tripped first.

C. Correct. According to the note at step 2.2 of the PK, NOTE: A vibration reading step change (calculated to be 10 mils) to Bearing 1 or 2 is indicative of a high pressure turbine blade being thrown. A mechanical failure of this magnitude should be seen on adjacent bearings. With power above P-9, the action is to trip the reactor.

D. Incorrect. First part is correct. Action is plausible as in OP AP-29, in multiple sections, there is a caution that states "Breaking vacuum after the Turbine trip will reduce turbine coastdown time. However, this should be done only as a last resort, since breaking vacuum with Turbine speed greater than 200 rpm could cause serious Turbine damage." A loss of bearing oil or thrown blade could be viewed as a time vacuum should be broken above 200 rpm...

Technical References: AR PK12-17, OP AP-29

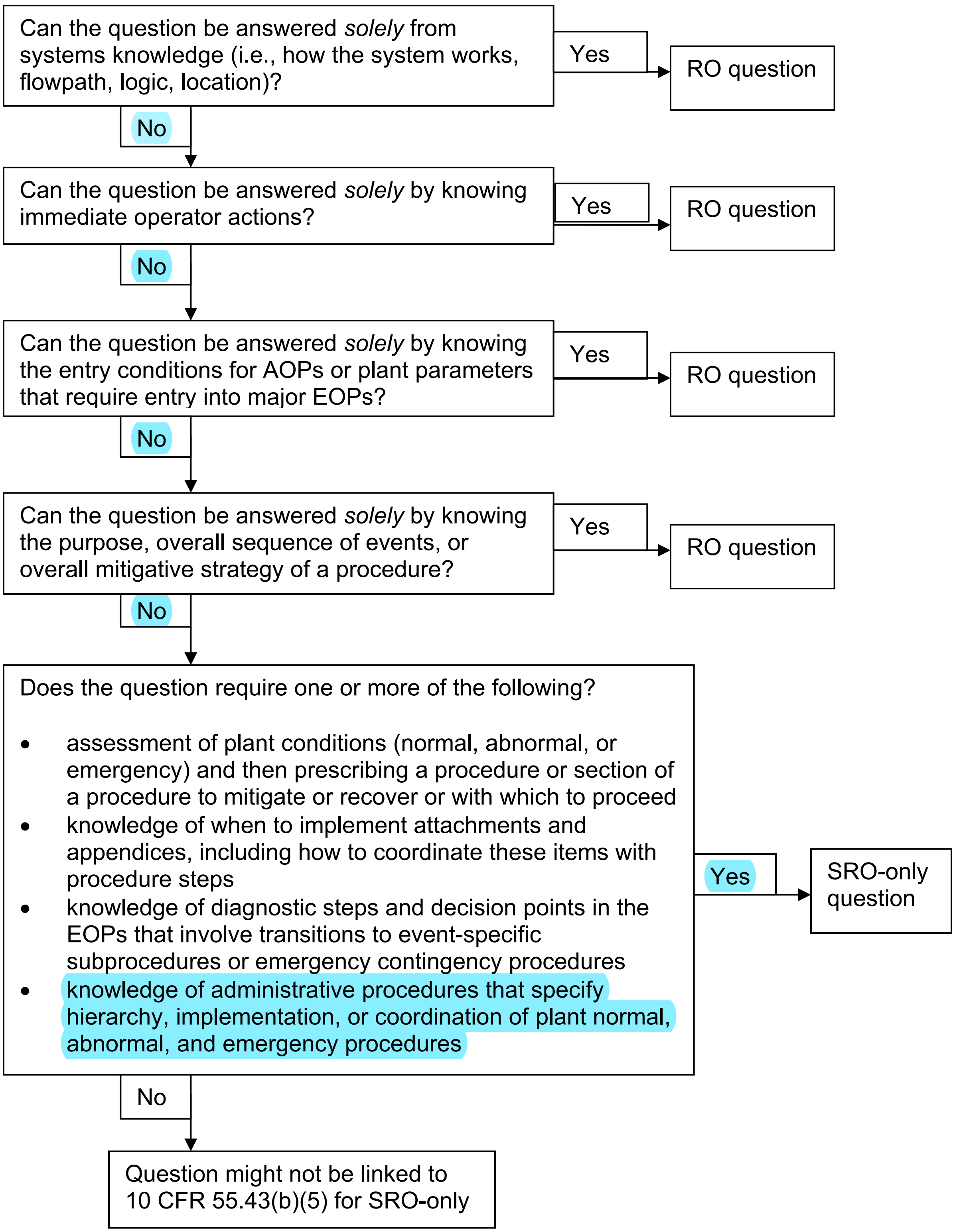
References to be provided to applicants during exam: None

Learning Objective: 3477AD - Given an abnormal condition, summarize the major actions of OP AP-29 to mitigate an event in progress.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	N
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 2.5

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



TURBINE SUPER-VISION

09/04/17
Effective Date

QUALITY RELATED

CONTAINS REACTOR TRIP CRITERIA

1. ALARM INPUT DESCRIPTION

INPUT	PRINTOUT/DETAILS	DEVICE	SETPOINT	STEP
1109	Turbine Supervisor Instru Drawer	YS-270		2.1
	INPUTS (see NOTE 1)		Alert / Danger	
	Vibration (> 600 RPM)		4 mils / 7 mils (see NOTE 2)	
	Eccentricity (< 600 RPM)		4.5 mils/5 mils	
	Differential Expansion			
	Rotor Long-Governor end		+320 mils/ +350 mils	
	Rotor Long-Generator end		+272 mils/ +351 mils	
	Rotor Short-Governor end		-95 mils/ -105 mils	
	Rotor Long-Generator end		-357 mils/ -475 mils	
	Rotor Position		±25 mils/ ±35 mils	
	Turbine Speed Zero		N/A	

NOTE 1: Also inputs to HMI YI-2001 on VB4.
NOTE 2: This is the "nominal" setpoint unless changed to accommodate plant conditions. Vibration setpoints at other than the "nominal" value should be noted on the Plant Abnormal Status Board.

2. OPERATOR ACTIONS

2.1 General Actions

2.1.1 Identify the alarm on the HMI Y1-2001 on VB4 as follows:

a. Ensure "Main Overview" screen is displayed on HMI.

b. Touch alarming point (in the bordered box).

c. Select "Bearing Charts" to trend the alarming bearing(s).

NOTE: Any subsequent alarms will be visible in the message window.

2.1.2 Validate the alarm as follows:

a. Check adjacent bearing indications (for vibration alarms).

b. Dispatch an Operator to check the following:

1. Bearing oil temperatures

2. Local Bently-Nevada panel on 140 ft or 85 ft Turbine Building (if necessary)

2.1.3 IF at any time a vibration indication is greater than or equal to 14 mils, THEN perform the following:

a. IF Reactor power is greater than P-9, THEN TRIP the Reactor and GO TO EOP E-0, "Reactor Trip or Safety Injection."

b. IF Reactor power is less than P-9, THEN TRIP the Main Turbine and GO TO OP AP-29, "Main Turbine Malfunction."

2.1.4 GO TO the appropriate section for the current alarm as follows:

- High Vibration - Section 2.2
- Differential Expansion - Section 2.3
- Rotor Position - Section 2.4
- Turbine Zero Speed - Section 2.5
- Not OK - Section 2.6

2.2 High Vibration

> **NOTE:** A vibration reading step change (calculated to be 10 mils) to bearings 1 and/or 2 is indicative of a high pressure turbine blade being thrown. A mechanical failure of this magnitude should be seen on adjacent bearings.

2.2.1 IF there has been a step change of 10 mils or greater on bearings 1 and/or 2,
THEN perform the following:

a. IF Reactor power is greater than P-9,
THEN TRIP the Reactor and GO TO EOP E-0, "Reactor Trip or Safety Injection."

b. IF Reactor power is less than P-9,
THEN TRIP the Main Turbine and GO TO OP AP-29, "Main Turbine Malfunction."

2.2.2 IF there is a valid vibration alarm locked in,
AND a Reactor/Turbine trip is NOT required,
THEN GO TO OP AP-29, "Main Turbine Malfunction."

2.2.3 Probable Causes

- Turbine passing through a critical frequency on a startup
- Damaged blading or bearing
- Change in seal oil temperature (bearing 11 is very sensitive to changes in generator seal oil temperature)
- Blockage of lube oil supply following a turbine maintenance outage (would be coincident with high bearing oil return or bearing metal temperatures)

2.3 Differential Expansion

2.3.1 IF at any time differential expansion danger value is reached or exceeded,
THEN perform the following:

a. IF Reactor power is greater than P-9,
THEN TRIP the Reactor and GO TO EOP E-0, "Reactor Trip or Safety Injection."

b. IF Reactor power is less than P-9,
THEN TRIP the Main Turbine and GO TO OP AP-29, "Main Turbine Malfunction."

2.3.2 IF a differential expansion point is in Alert,
THEN GO TO OP AP-29, "Main Turbine Malfunction."

2.3.3 Probable Causes

- Heatup too rapid

2.4 Rotor Position

2.4.1 Observe and trend rotor position.

2.4.2 GO TO OP AP-29, "Main Turbine Malfunction."

2.4.3 Probable Causes

- Thrust bearing wear or failure
- Load changes
- Grid disturbances

2.5 Turbine Zero Speed

NOTE:	The turning gear will automatically engage 60 seconds after reaching zero speed on both indicators. If the indicators do not register any shaft rotation for 4 seconds after the turning gear attempts to engage, this alarm pop-up will occur.

2.5.1 Check turbine shaft rotation is no longer required.

2.5.2 IF turbine shaft rotation is required,
THEN engage turning gear PER OP C-3:IV, "Main Unit Turbine - Turbine Shutdown."

2.5.3 Reset the alarm PER Section 2.7.

2.5.4 Probable Causes

- Turning gear failed to engage
- Speed probe failure

2.6 Not OK

2.6.1 IF the HMI YI-2001 pop-up display message contains "Not OK",
THEN contact Maintenance.

2.6.2 Probable Causes

- Channel failure

2.7 <u>Reset Alarms</u>

> **<u>NOTE</u>:** The following steps will acknowledge and reset all alarms currently in the system, including those at the TSI Panel on 140 ft and the MFW Pump panels on 85 ft in the Turbine Building.

2.7.1 Select "Main Overview" on the HMI.

2.7.2 Acknowledge all alarms in the system as follows:

a. Select the "Acknowledge" button.

OR

b. Right-click anywhere in the alarm list and navigate to "Ack All" or "Ack Page."

2.7.3 Perform a "Rack Reset" as follows:

a. Select the "Rack Reset" button.

b. Ensure the expected number of alarms is present in the pop-up box.

c. Select "MT Rack Reset" in the pop-up box.

3. AUTOMATIC ACTIONS

None

4. REFERENCES

None

5. LOGIC DIAGRAM

YS 270 ⟶ ALARM

APPENDIX 3.2
Turbine Lube Oil System (PK12-16)

1. PROBABLE CAUSE

1.1 High or low bearing oil header pressure. **(Input 719)**

1.2 Emergency bearing oil pump running. **(Input 724)**

1.3 Emergency bearing oil pump UV or OC. **(Input 720, 723)**

1.4 Bearing oil pump UV or OC. **(Input 726)**

1.5 Lube oil reservoir level high or low. **(Input 721, 722)**

1.6 Main Turbine lube oil cooler high temperature (could be due to loss of service cooling water). **(Input 1248)**

2. AUTOMATIC ACTIONS

2.1 Start of AC or DC bearing oil pump.

3. OPERATOR ACTIONS for Input 719 (PPC Lube Oil Pressure Alarm)

3.1 Determine if pressure is high or low.

3.2 IF pressure is low,
THEN perform the following:

3.2.1 Start the AC and/or DC bearing oil pump.

CAUTION:	Breaking vacuum after the Turbine trip will reduce turbine coastdown time. However, this should be done only as a last resort, since breaking vacuum with Turbine speed greater than 200 rpm could cause serious Turbine damage.

3.2.2 IF bearing oil header pressure remains low,
THEN stop the turbine shaft as follows:

a. IF Reactor power is greater than P-9,
THEN TRIP the Reactor and GO TO EOP E-0, "Reactor Trip or Safety Injection."

b. IF Reactor power is less than P-9, Turbine Trip without Reactor Trip,
THEN TRIP the Main Turbine and GO TO step 9 of this procedure.

3.3 IF pressure is high,
THEN check lube oil cooler outlet temperature to see if it is low.

3. OPERATOR ACTIONS for Input 611 (Thrust or Journal Bearing Problem)

3.1 Check high bearing temperature(s) on the PPC by entering, "MAINTURBINEBRG" in the window name area, and check locally at turbine generator (oil return temp). Exciter bearing #11 has metal temperature indication only.

3.2 Check adequate lube oil return flow from each bearing.

CAUTION:	Breaking vacuum after the Turbine trip will reduce turbine coastdown time. However, this should be done only as a last resort, since breaking vacuum with Turbine speed greater than 200 rpm could cause serious Turbine damage.

3.3 IF any of the following conditions exist:

- Loss of oil return flow observed at any bearing.
- One or more bearing oil return temperature is verified to be above 180°F (no oil return temperature indication on Bearing #11).
- One or more bearing metal temperature reads greater than 225°F on the PPC, THEN perform the following:

3.3.1 IF Reactor power is greater than P-9, THEN TRIP the Reactor and GO TO EOP E-0, "Reactor Trip or Safety Injection."

3.3.2 IF Reactor power is less than P-9, Turbine Trip without Reactor Trip, THEN TRIP the Main Turbine and GO TO step 9 of this procedure.

3.4 IF bearing oil or metal temperatures are between the alarm and manual trip setpoints, THEN:

3.4.1 Check turbine bearing oil pressure between 15 and 20 psig.

3.4.2 Check lube oil supply temperature to the bearings between 110°F and 120°F.

3.4.3 IF temperature is above 120°F, THEN check the following:

a. Proper operation of TCV-15. Select manual control and open as needed.

b. Adequate SCW supply to inservice lube oil cooler. Normal SCW supply temperature. Vent lube oil cooler (water side) as needed.

c. Correct alignment of lube oil coolers.

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	3
G2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	**Group #**	1
	K/A #	G2.1.7
	Rating	4.7

Question 19

GIVEN:

- The crew is performing the diagnostic steps of EOP E-0, Reactor Trip or Safety Injection
- RCS temperature is currently 315°F and rising after an initial decrease to 230°F
- RCS pressure 1750 psig and rising
- Pressurizer level 15% and rising
- Steam Generator Narrow Range levels:
  - o is 22%, rising slowly
  - o is 0%, stable
  - o is 18%, stable
  - o is 23%, rising slowly
- Steam Generator pressures:
  - o is 820 psig, rising
  - o is 0 psig, stable
  - o is 900 psig, stable
  - o is 890 psig, rising

What procedure should the crew transition to next?

A. EOP E-1.1, Safety Injection Termination

B. EOP E-2, Faulted Steam Generator Isolation

C. EOP FR-H.5, Response to Steam Generator Low Level

D. EOP FR-P.1, Response to Imminent Pressurized Thermal Shock Condition

Proposed Answer: B. EOP E-2, Faulted Steam Generator Isolation

Explanation:

SRO must evaluate the current conditions and determine which procedure entry is appropriate. Based on conditions, a RED challenge to RCS Integrity had existed, but according to rules of EOP usage, does not need to be addressed. Also, SI termination criteria is met, however, a faulted S/G, although completely depressurized, must be addressed prior to terminating SI.

A. Incorrect. A transition to E-2 must be made based on indication that Steam Generator 1-2 is faulted. Plausible because plant conditions indicate the MSIVs are closed (based on SG pressure indications), which is a required action for isolation and SI termination criteria are met.
B. Correct. Based on SG 1-2 completely depressurized, a transition from EOP E-0, to EOP E-2 is required.
C. Incorrect. Plausible as the criteria exists for EOP H.5 entry, however, a YELLOW path procedure does not take precedence over EOP E-2.
D. Incorrect. Plausible because a MAGENTA path on RCS Integrity CSF existed, however, according to the rules of usage, if the condition has cleared prior to implementation, the FRG need not be entered.

Technical References: F-0, F-0.4, EOP E-0

References to be provided to applicants during exam: None

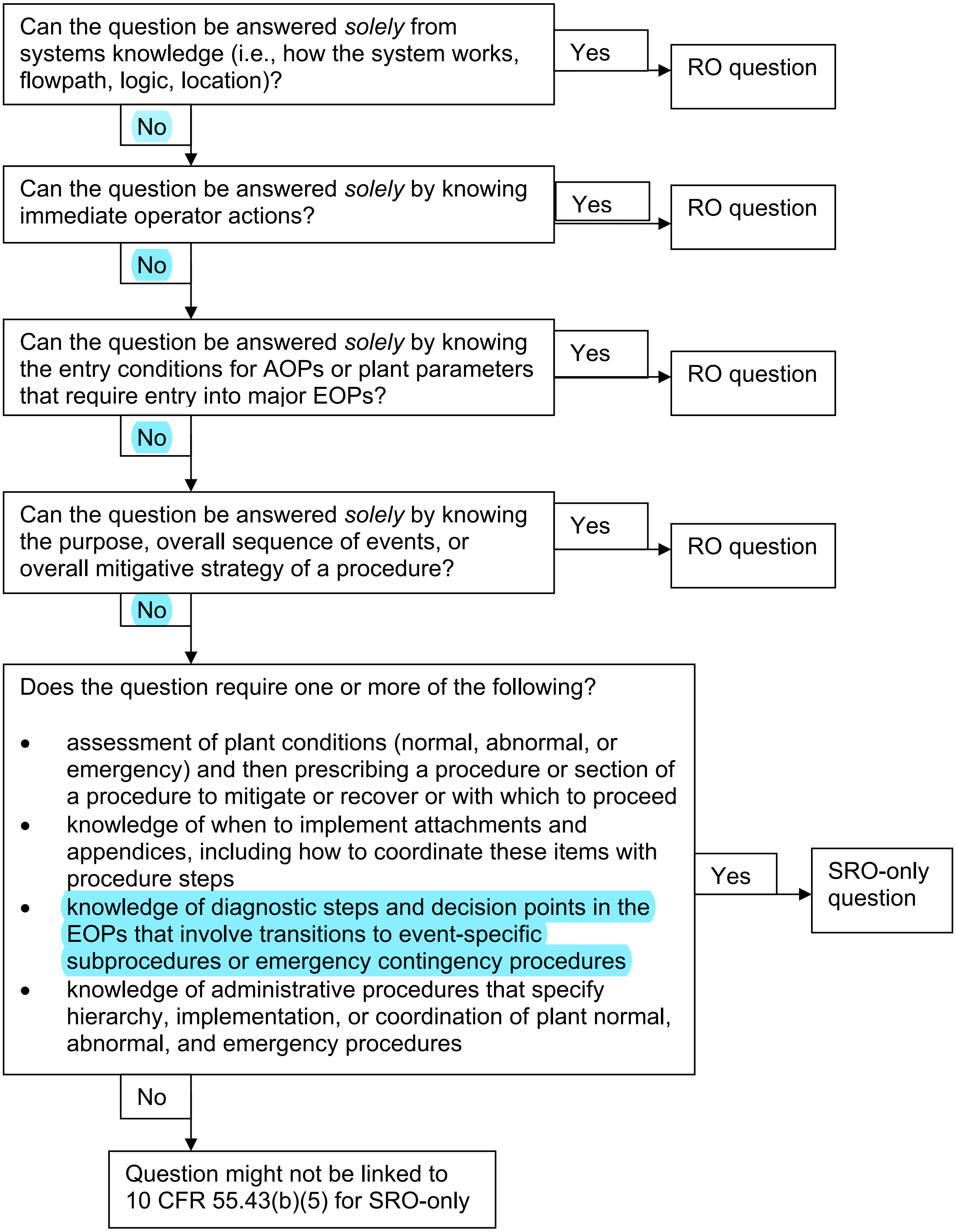
Learning Objective: 38107 - Apply the Rules of Usage in EOPs for the CSFSTs and FRGs, including: the six status trees - the priority of use of the status trees - the priority of use of the color of each CSF - when to monitor and/or implement the CSFSTs and FRGs

Question Source:	Bank #77 L181	Y
(note changes; attach parent)	Modified Bank	
	New	
	Past NRC Exam DCPP 03/2020	Y
Question History:	Last Two NRC Exams	Y
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 3.0

Question 19

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Examination Outline Cross-Reference	**Level**	SRO
	Tier #	2
039 G2.1.7 – Main and Reheat Steam: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	**Group #**	1
	K/A #	039 G2.1.7
	Rating	4.7

Question 77 PARENT QUESTION

GIVEN:

- The crew is performing the diagnostic steps of EOP E-0, Reactor Trip or Safety Injection
- RCS temperature is currently 315°F and rising after an initial decrease to 230°F
- RCS pressure 1750 psig and rising
- Pressurizer level 15% and rising
- Steam Generator Narrow Range levels:
  - 1-1 is 22%, rising slowly
  - 1-2 is 0%, stable
  - 1-3 is 18%, stable
  - 1-4 is 23%, rising slowly
- Steam Generator pressures:
  - 1-1 is 820 psig, rising
  - 1-2 is 0 psig, stable
  - 1-3 is 900 psig, stable
  - 1-4 is 890 psig, rising

What procedure should the crew transition to next?

A. EOP E-1.1, Safety Injection Termination

B. EOP E-2, Faulted Steam Generator Isolation

C. EOP FR-H.5, Response to Steam Generator Low Level

D. EOP FR-P.1, Response to Imminent Pressurized Thermal Shock Condition

Proposed Answer: B.EOP E-2, Faulted Steam Generator Isolation

Explanation:

SRO must evaluate the current conditions and determine which procedure entry is appropriate. Based on conditions, a RED challenge to RCS Integrity had existed, but according to rules of EOP usage, does not need to be addressed. Also, SI termination criteria is met, however, a faulted S/G must be addressed prior to terminating SI>

A. Incorrect. Although conditions support SI termination, a transition to E-2 will be made based on Steam Generator 1-2 completely depressurized.
B. Correct. A transition from E-0 is required to perform the faulted steam generator isolation.
C. Incorrect. Conditions met for entry but H.5 is a YELLOW path and not addressed at this time.
D. Incorrect. The initial decrease was to a temperature that made the status tree for RCS integrity Red, however, the status is currently Green. The condition has cleared and does need to be addressed.

Technical References: F-0, E-0

References to be provided to applicants during exam: None

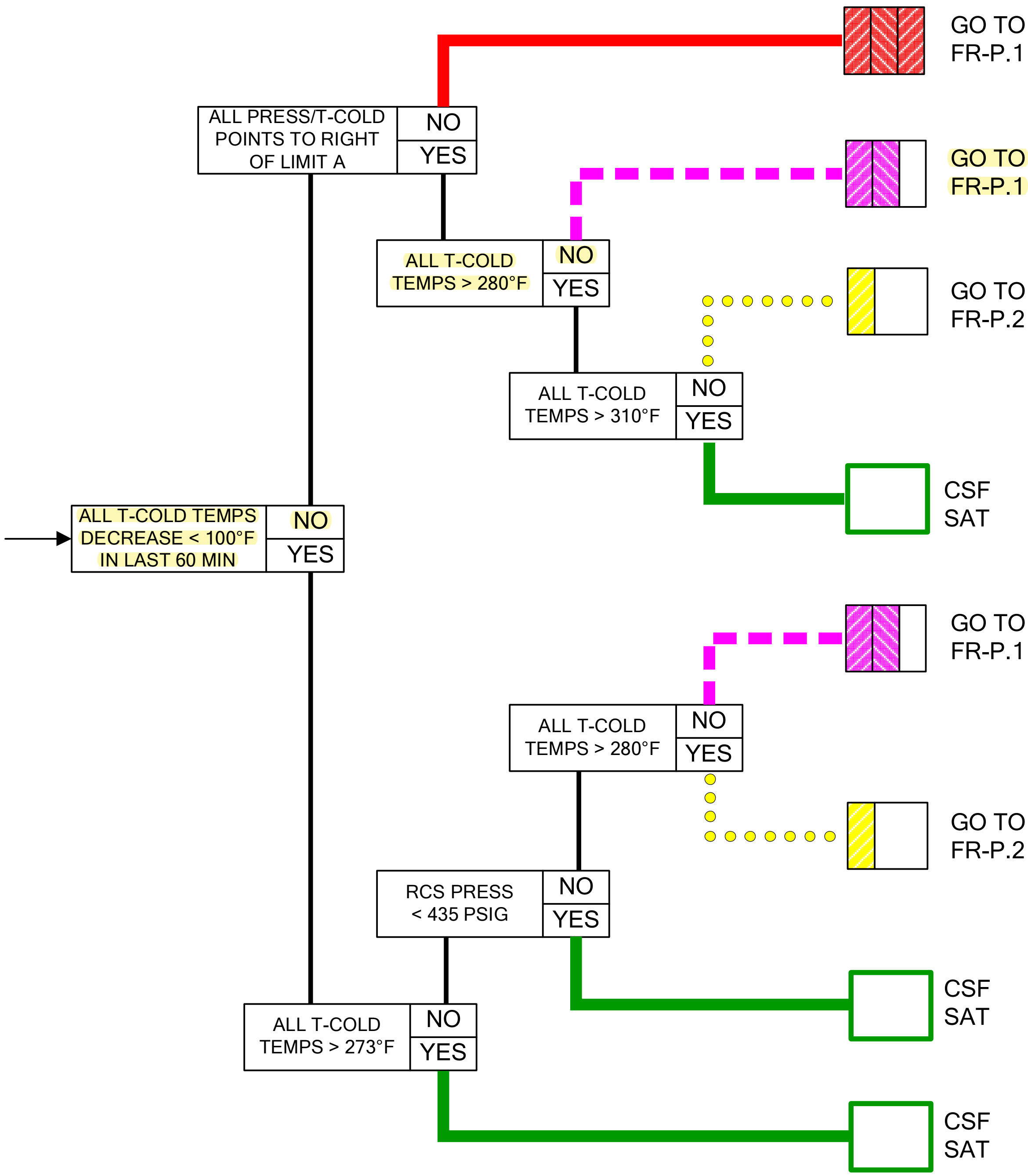
Learning Objective: 3552 - Given initial conditions, assumptions, and symptoms, determine the correct Emergency Operating Procedure to be used to mitigate an operational event

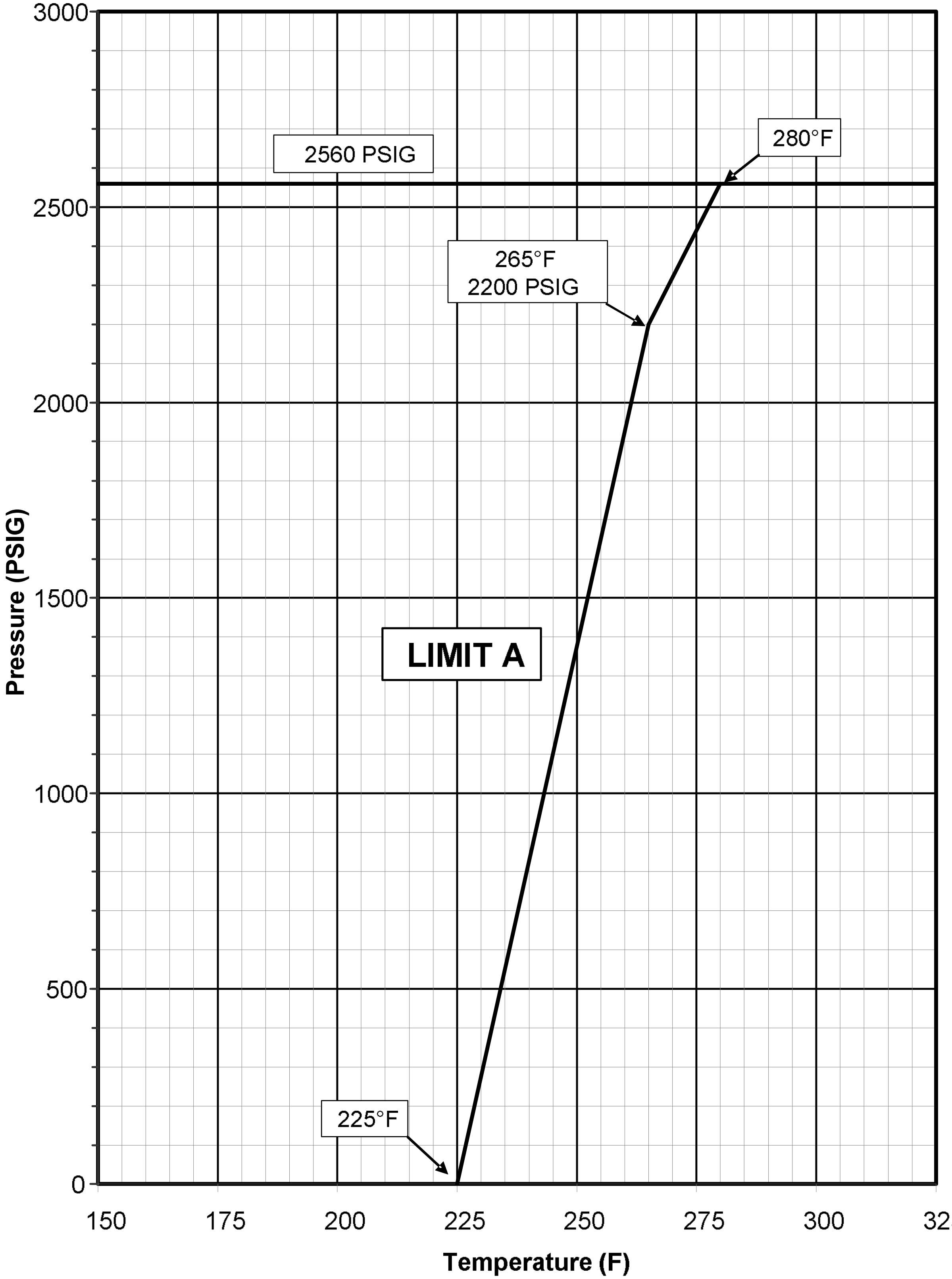
Question Source:	Bank #85 DCPP L081 01/2010	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 01/2010	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 3.0

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
◊ **7. CHECK PZR SAFETY VLVS, PORVs, And Spray Vlvs: (Continued)**	
c. Block vlvs - ALL BLOCK VALVES OPEN	c. IF Power is available, THEN Open all block vlv(s) unless closed to isolate an open or failed PORV.
d. Normal PZR Spray Control Vlvs - CLOSED	d. IF PZR pressure LESS THAN 2260 PSIG, THEN Ensure proper valve response OR Close PZR Spray Control Vlvs. IF Valves CANNOT be closed, THEN Stop RCP 1 AND RCP 2.
◊ **8. CHECK If RCPs Should Be STOPPED:**	
a. Check WR RCS pressure LESS THAN 1300 PSIG	a. GO TO step 9.
b. Ensure AT LEAST one ECCS CCP or SI Pp running, delivering flow	b. GO TO step 9.
c. Stop All RCPs	
d. Maintain RCP Seal Injection between 8 GPM and 13 GPM by throttling FCV-128	d. IMPLEMENT OP AP-28, REACTOR COOLANT PUMP MALFUNCTION, for Loss of Seal Injection.
◊ **9. CHECK S/Gs - NOT FAULTED:**	GO TO EOP E-2, FAULTED STEAM GENERATOR ISOLATION.
a. NO S/G Pressure lowering in an uncontrolled manner	
b. NO S/G Completely depressurized	

F-0.4 RCS INTEGRITY




2560 PSIG
280°F
265°F
2200 PSIG
LIMIT A
225°F
3000
2500
2000
1500
1000
500
0
Pressure (PSIG)
150
175
200
225
250
275
300
325
Temperature (F)

3.7 Once implemented, a RED or MAGENTA Function Restoration Procedure must be performed to the point of the defined transition regardless of whether or not the RED or MAGENTA condition has been cleared.

3.8 IF a Functional Restoration Guideline is in progress due to a severe challenge (MAGENTA PATH) AND the CSF Status Tree goes to an extreme challenge (RED PATH)
AND references the SAME guideline,
THEN the operator should continue in the guideline from the current step since the corrective actions are the same regardless of the severity of the challenge.

3.9 IF an extreme challenge (RED PATH) or a severe challenge (MAGENTA PATH) is diagnosed
AND subsequently clears before the status trees are being implemented or during the performance of a higher priority function restoration,
THEN that challenge is to be considered satisfied.

3.10 IF extreme or severe challenges exist
OR plant conditions are changing rapidly,
THEN critical safety function status trees shall be monitored continuously.

3.11 IF extreme or severe challenges do NOT exist
AND plant conditions are NOT changing rapidly,
THEN critical safety function status trees shall be monitored every 10 to 20 minutes.

3.12 IF conditions for transfer to cold leg recirculation exist,
THEN critical safety function status trees shall NOT be implemented until directed by EOP E-1.3, "Transfer to Cold Leg Recirculation."

4. SAFETY PARAMETERS DISPLAY SYSTEM

4.1 F-0.4 RCS INTEGRITY

4.1.1 In the event of an unexpectedly severe RCS cooldown (i.e., pressurized thermal shock) or an unexpected overpressure condition at low temperature during a controlled cooldown (i.e., cold overpressure), a RED or MAGENTA condition may come in temporarily and subsequently stabilize in a YELLOW or GREEN condition. If this occurs, the operator is not required to perform the actions of FR-P.1 since adequate time was not allowed for thermal stresses to affect the integrity of the vessel wall.

3. RULES OF USAGE

3.1 The Critical Safety Function Status Trees shall be monitored in the following order of priority:

3.1.1 Subcriticality using Status Tree F-0.1

3.1.2 Core Cooling using Status Tree F-0.2

3.1.3 Heat Sink using Status Tree F-0.3

3.1.4 RCS Integrity using Status Tree F-0.4

3.1.5 Containment using Status Tree F-0.5

3.1.6 Inventory: Using Status Tree F-0.6 [T35691]

3.2 IF an extreme challenge (RED PATH) is diagnosed,
THEN the operator shall IMMEDIATELY stop procedure in effect and initiate functional restoration to restore the critical safety function under extreme challenge.

3.3 IF a severe challenge (MAGENTA PATH) is diagnosed,
THEN the operator shall continue to check the status of all remaining critical safety functions.

IF no extreme challenges exist,
THEN the operator shall stop procedure in effect and initiate functional restoration to restore the highest priority critical safety function under severe challenge.

3.4 IF a not satisfied condition (YELLOW PATH) is diagnosed,
THEN the operator shall continue to check the status of all remaining critical safety functions.

IF no extreme or severe challenges exist,
THEN it is the operator's option to continue optimal recovery procedures or to initiate functional restoration of the affected critical safety function challenge.

3.4.1 Once implemented, a YELLOW PATH Function Restoration Procedure may be terminated based on operator judgment.

3.4.2 During performance of a YELLOW PATH Functional Restoration Procedure, the continuous actions and foldout page items of the E or ECA procedure in effect are still applicable and should be monitored by the operator.

3.5 IF a satisfied condition (GREEN PATH) is diagnosed,
THEN no challenge exists for the affected critical safety function.
The operator shall continue to check the status of all remaining critical safety functions.

3.6 IF during function restoration to address a critical safety function challenge, a higher priority challenge is diagnosed,
THEN the operator should suspend the ongoing response and initiate function restoration to address the higher priority critical safety function challenge.

Examination Outline Cross-Reference		**Level**	SRO
		Tier #	3
G2.1.42 Knowledge of new and spent fuel movement procedures. (SRO Only)		**Group #**	1
		K/A #	G2.1.42
		Rating	3.4

Question 20

In accordance with OP L-6, Cold Shutdown/Refueling, and OP B-8DS1, Core Unloading, which of the following is the first CORE ALTERATION activity requiring the presence of the Refueling SRO in Containment?

A. Lifting the reactor vessel head

B. Unlatching RCCAs

C. Lifting the upper internals

D. Moving the first fuel assembly

Proposed Answer: B. Unlatching RCCAs

Explanation:
CORE ALTERATION - shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
A. Incorrect. Plausible as this occurs prior to unlatching RCCAs, but not a core alteration
B. Correct. Per OP B-8DS1, the Refueling SRO is responsible for supervising Core Alterations. RCCA unlatching and moving of fuel assemblies are core alterations.
Unlatching is performed prior to removal of the fuel assembly.
C. Incorrect. Plausible as it is an action that is taken but after unlatching RCCAs
D. Incorrect. Plausible as it is a core alteration but performed after unlatching RCCAs

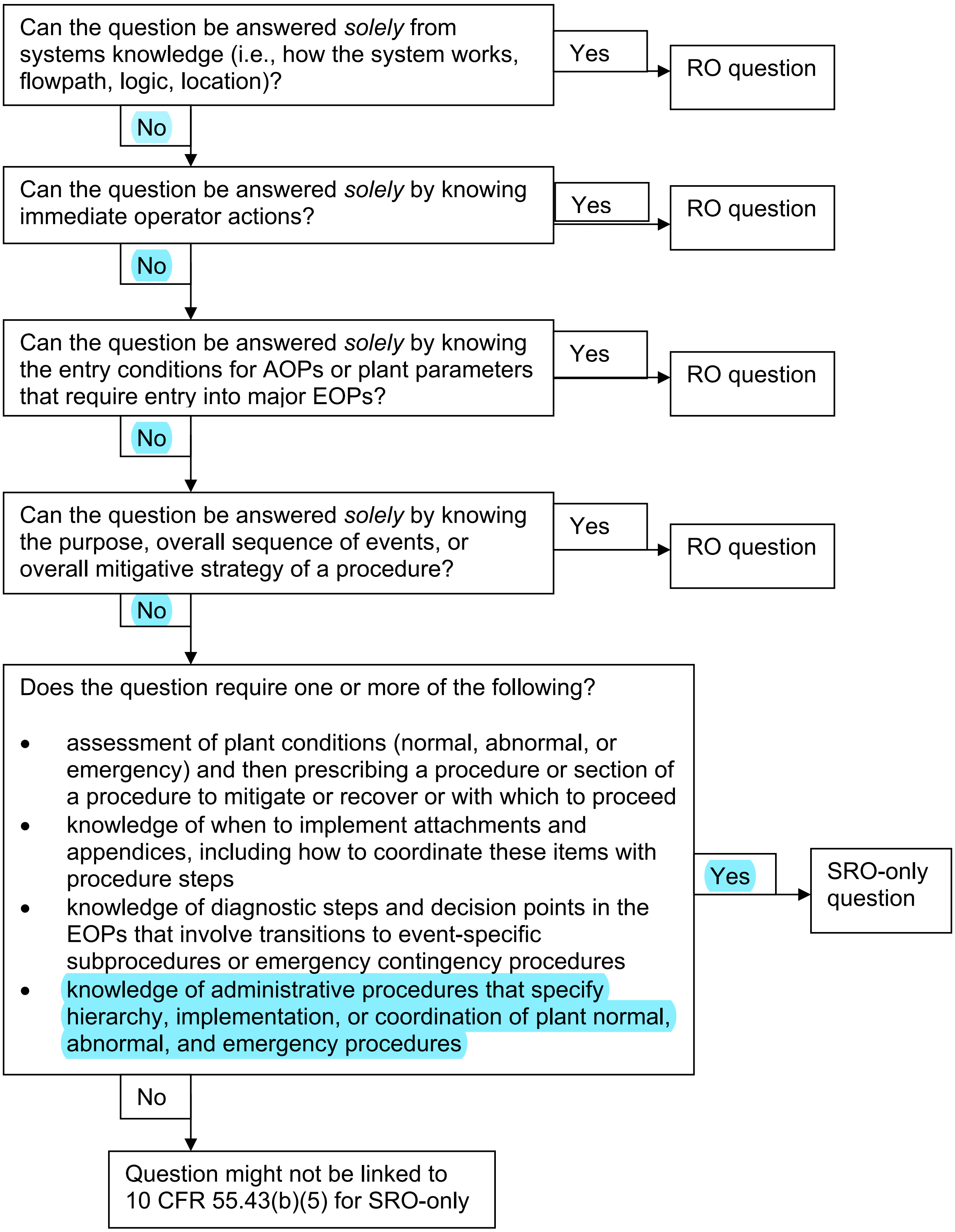
Technical References: OP L-6, OP B-8DS1
References to be provided to applicants during exam: none
Learning Objective: 6497 - State the responsibilities and duties of Refueling SRO

Question Source:	Bank #95 L161	X
(note changes; attach parent)	Modified Bank	
	New	
	Past NRC Exam DCPP 10/16	Y
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

10CFR Part 55 Content: 55.43.7

Difficulty: 2.5

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Examination Outline Cross-Reference	Level	SRO
G2.1.41 – Knowledge of refueling process	**Tier #**	3
	Group #	1
	K/A #	G2.1.41
	Rating	3.7

Question 95 PARENT QUESTION

Which of the following would be the FIRST core alteration activity requiring direct supervision by the Refueling SRO?

A. Lifting the reactor vessel head

B. Moving the first fuel assembly

C. Lifting the upper internals

D. Unlatching RCCAs

Proposed Answer: D. Unlatching RCCAs

Explanation:
Plausible, all actions take place during fuel movement

A. Incorrect. Not a core alteration
B. Incorrect. This is a core alteration, but not the first one listed
C. Incorrect. Not a core alteration
D. Correct. This is a core alteration and done prior to moving any fuel assemblies

Technical References: OP L-6, OP B-8DS1
References to be provided to applicants during exam: none
Learning Objective: 6497 - State the responsibilities and duties of Refueling SRO

Question Source:	Bank #96 DCPP NRC Exam 02/2005	
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Past NRC Exam	Yes
	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.6	

SRO is responsible for supervision of refueling activities.

Difficulty: 2.5

2.2 This procedure ensures fuel in transit can always be lowered into a safe position quickly.

2.2.1 Fuel "in transit" includes latched in Containment, in the transfer system, and latched in the Spent Fuel Pool.

2.2.2 A "safe position" includes fully lowered into a Spent Fuel Pool cell, in the upender and lowered with the upender on the containment side, and fully lowered into the core.

3. RESPONSIBILITIES

3.1 Shift Foreman is responsible for overseeing the operation of the plant and plant equipment.[T35644]

3.2 The Refueling SRO is responsible for coordinating and supervising the following activities:

- Direct supervision of CORE ALTERATION activities with no concurrent duties.
- All fuel handling operations.
- Safe and orderly evacuation of the refueling crew in the event of a high radiation alarm at a refueling station.
- Determining the cause of high radiation alarms.
- Determining that fuel handling personnel are properly qualified for their duty stations.
- The Refueling SRO may delegate supervisory responsibilities at the other duty stations outside of containment to a designated operations representative.
- Notifying the Shift Manager following any human performance error or near miss.
- Evaluate safe and reliable fuel handling equipment performance and need for corrective action

3.3 The Fuel Handling Building Supervisor (FHBS) reports directly to the Refueling SRO and is responsible for providing oversight of all fuel handing building (FHB) activities including, but not limited to the following:

- Provide oversight to all aspects of fuel handling activities in the SFP (monitoring load and elevation indication, clear travel paths of the fuel and unloaded spent fuel handling tool, location verifications) with no concurrent duties
- Ensures a safe and orderly evacuation of the fuel handling building in the event of a FHB evacuation alarm
- Ensures the fuel handling team is utilizing human performance standards.
- Ensure the safe handling of nuclear fuel
- Shall avoid any activities that distract the FHBS from maintaining an oversight role
- Shall provide an independent check of fuel locations following the spotter and fuel handler concurrent verification
- Shall report any equipment issues to the Refueling SRO
- Shall report any human performance issues to the Refueling SRO
- Has overall accountability for all Spent Fuel Pool activities

NOTE:	The period of time between disconnecting the last two incore thermocouples and lifting the Reactor Vessel Head is an exception to AD8.DC55, Attachment 4, "Unit 1 Outage Safety Checklist - MODE 6 RCS Level Greater Than or Equal to 111'." This time should be minimized. REFER TO the Outage Safety Plan.

m. Just prior to lifting the Reactor Vessel Head, authorize maintenance to disconnect the last two incore thermocouples. _____/_____/_____

n. Authorize maintenance to remove the Reactor Vessel Head. _____/_____/_____

NOTE:	The refueling cavity should be filled continuously until level is >130 ft, unless reactor cavity seal leakage or other problem requires draindown.

o. WHEN the Reactor Vessel head is clear of the guide pins,
THEN direct Operations personnel to commence filling the refueling cavity PER OP B-2:II. _____/_____/_____

p. WHEN the Reactor Vessel Head is placed on its stand,
THEN align Control Room ventilation per SFM direction. _____/_____/_____

NOTE:	Once the Refueling Cavity is filled, makeup to the cavity should be performed using OP B-2:II, "RHR - Filling the Refueling Cavity."

q. IF PCV-135 was bypassed to increase flow for RCS cleanup,
THEN return to normal PER OP B-1A:XVII, "Enhanced RHR Letdown Flow for RCS Cleanup." _____/_____/_____

r. OPEN the RHR-1-8703 equalizing valve that was closed in step 6.1.1g.2.

- RHR-1-1005 _____/_____/_____
- RHR-1-933 _____/_____/_____

NOTE:	Unlatching RCCA drive shafts IS a CORE ALTERATION.

s. Perform the following for preparation and authorization of RCCA unlatching:

1. Complete the OP L-0 Transition checklist for MODE 6 to CORE ALTERATIONS. _____/_____/_____

Examination Outline Cross-Reference		**Level**	SRO
		Tier #	3
G2.2.45 Ability to determine and/or interpret TS with action statements of greater than 1 hour. (SRO Only)		**Group #**	2
		K/A #	G2.2.45
		Rating	4.7

Question 21

A pump is declared inoperable and the Shift Foreman enters a 72 hour LCO ACTION, applicable in MODEs 1 through 4.

8 hours later, the second pump in the system is declared inoperable and the Shift Foreman determines LCO 3.0.3 applies.

3 hours later, the second pump is restored to OPERABLE status and the Shift Foreman exits LCO 3.0.3

The maximum time remaining to return the first pump to OPERABLE status is ______ hours.

A. 61

B. 64

C. 72

D. 76

Proposed Answer: A. 61

Explanation:
LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:
a. MODE 3 within 7 hours;
b. MODE 4 within 13 hours; and
c. MODE 5 within 37 hours.
Per Section 1.3 While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited, and *operation continued in accordance with Condition A*. In this case, from the original 72 hours, 11 hours have expired and 61 hours remain.
NOTE: Per section 1.3 of Technical Specifications, To apply a Completion Time extension, two criteria must first be met. The subsequent inoperability:
a. Must exist concurrent with the first inoperability; and
b. Must remain inoperable or not within limits **after the first inoperability is resolved.**

A. Correct. From the original LCO entry, a total of 11 hours have expired. Per Section 1.3, 61 hours remain.
B. Incorrect. This is plausible if only the first 8 hours are used and its thought the time in LCO 3.0.3 did not count (the tracking of the LCO stopped when entering 3.0.3 but restarted at that point when 3.0.3 exited)

C. Incorrect. Plausible if its thought a new completion time begins when LCO 3.0.3 is exited, then a new 72 hours would apply.
D. Incorrect. This is plausible if its thought the remaining time to MODE 3 in LCO 3.0.3 can be added to a new 72 hour completion time. Per section 1.3, no extension applies.

Technical References: LCO 3.0.3, Tech Specs section 1.3

References to be provided to applicants during exam: None

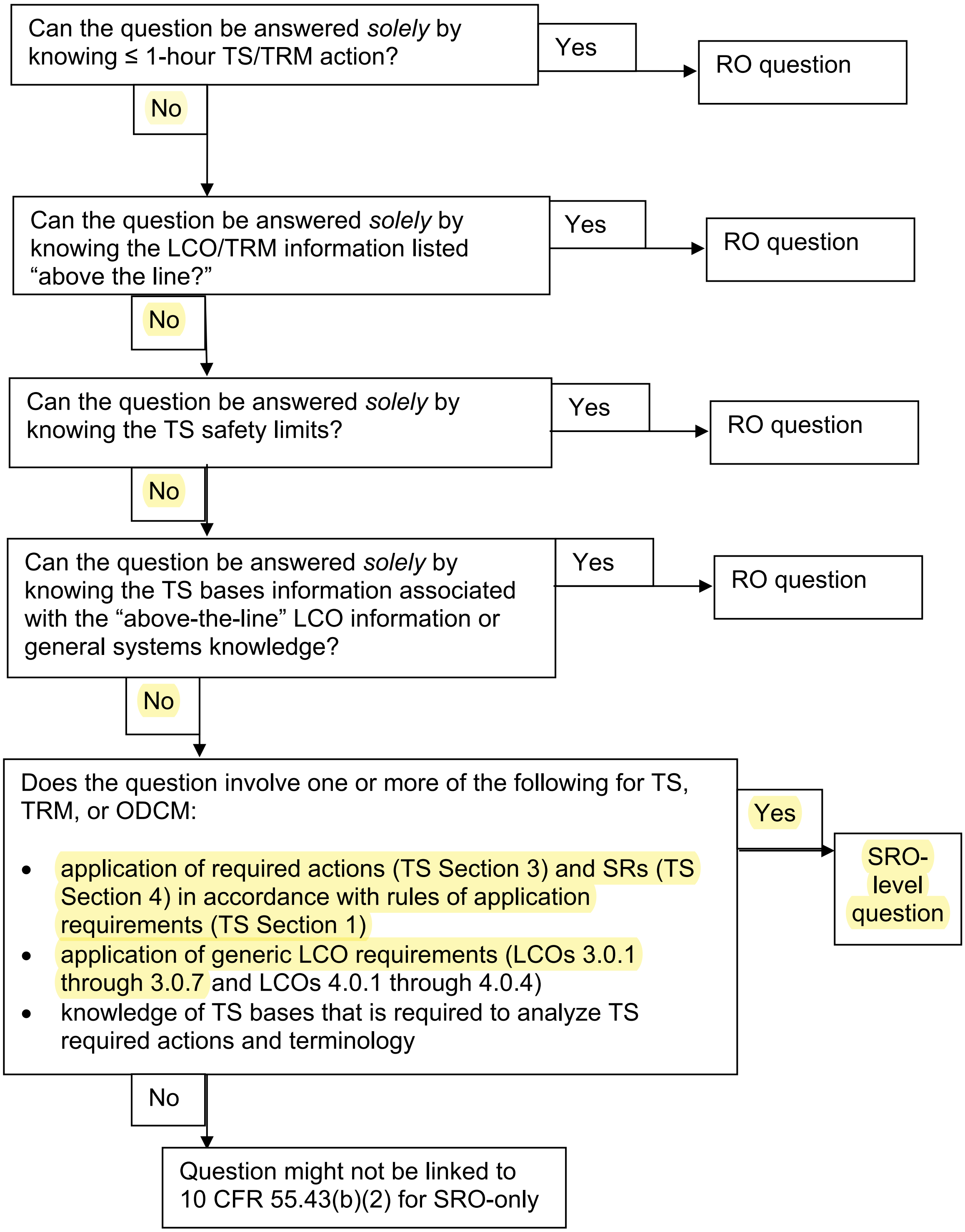
Learning Objective: 9700 - Explain LCO 3.0 & SR 3.0 general requirements

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	N
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.2	

Difficulty: 2.5

Question 21

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



c. *Facility Licensee Procedures Required to Obtain Authority for Design and Operating Changes in the Facility [10 CFR 55.43(b)(3)]*

Some examples of SRO-only examination items for this topic include the following:

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, and LCO 3.0.8.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.
LCO 3.0.3	When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in: a. MODE 3 within 7 hours; b. MODE 4 within 13 hours; and c. MODE 5 within 37 hours. Exceptions to this Specification are stated in the individual Specifications. Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required. LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.
LCO 3.0.4	When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made: a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time; b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or c. When an allowance is stated in the individual value, parameter, or other Specification. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

(continued)

Examination Outline Cross-Reference

G2.2.41 Ability to obtain and interpret station electrical and mechanical drawings. (reference potential)

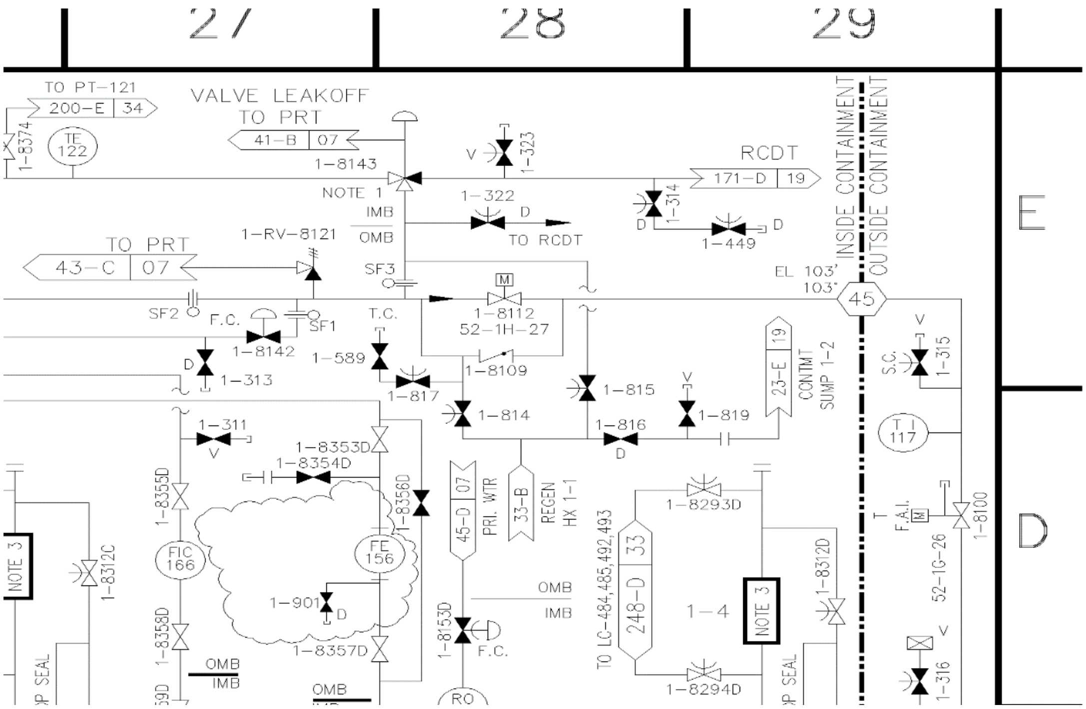
Level	SRO
Tier #	3
Group #	2
K/A #	G2.2.41
Rating	3.9

Question 22

Unit 1 is at 100% power.

A 30 gpm leak develops at the inlet of RCP Seal Water Return Stop valve, CVCS-8100.

1. The crew stopped the leakage in 12 minutes by closing ________.
2. EAL classification _______ required.



A. 1. CVCS-8100
   2. is

B. 1. CVCS-8100
   2. is NOT

C. 1. CVCS-8112
   2. is

D. 1. CVCS-8112
   2. is NOT

Proposed Answer: D. 1. CVCS-8112
2. is NOT

Explanation:
A review of the OVID shows that if the leak is upstream (inlet) of 8100, it can only be isolated by closing 8112.

Per the EAL appendix: RCS leakage outside of the containment that is not considered identified or unidentified leakage per Technical Specifications includes leakage via interfacing systems such as RCS to the Component Cooling Water, or systems that directly see RCS pressure outside containment such as **Chemical & Volume Control System**, Nuclear Sampling System and Residual Heat Removal system (when in the shutdown cooling mode). **If the leak is isolated, the RCS barrier was never lost**.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. An emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated).

A. Incorrect. Closing 8100 will not isolate the leak. Plausible because it is in the flow stream and if the flowpath or inidication is misread, closing 8100 would be the answer. The flow is from 8112 thru 8100. With the leak on the inlet of 8100, 8112 must be closed. Because the leakage is into a system and isolated by closing 8112, EAL classification is not required. Plausible to classify based on the fact it is RCS that is leaking at greater than 25 gpm. If it took longer than 15 minutes to isolate, it would be classifiable.
B. Incorrect. Closing 8100 will not isolate the leak. The flow is from 8112 thru 8100. With the leak on the inlet of 8100, 8112 must be closed. Second part is correct. Because the leakage is into a system and isolated by closing 8112 and in less than 15 mintutes, EAL classification is not required.
C. Incorrect. First part is correct. Closing 8100 will not isolate the leak. The flow is from 8112 thru 8100. With the leak on the inlet of 8100, 8112 must be closed. Because the leakage is into a system and isolated by closing 8112, EAL classification is not required. Plausible because it is RCS that is leaking.
D. Correct. Closing 8100 will not isolate the leak. The flow is from 8112 thru 8100. With the leak on the inlet of 8100, 8112 must be closed. Because the leakage is into a system and isolated by closing 8112, EAL classification is not required. Also, while the relief valve, 8121 may lift, because its not a malfunction of the valve, it is not classifiable.

Technical References: OVID 106708 sheet 2, EAL Appendix D, page 188

References to be provided to applicants during exam: Drawing in the stem

Learning Objective: Discuss operator behaviors and practices related to the operator fundamental of closely monitoring plant indications and conditions. (56218)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #87 L121	Y
	New	
	Past NRC Exam DCPP 08/14	Y
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.2	

Difficulty: 3

Question 22

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)

Can the question be answered *solely* from systems knowledge (i.e., how the system works, flowpath, logic, location)?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing the entry conditions for AOPs or plant parameters that require entry into major EOPs?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No ↓

Does the question require one or more of the following?

- assessment of plant conditions (normal, abnormal, or emergency) and then prescribing a procedure or section of a procedure to mitigate or recover or with which to proceed
- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific subprocedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No ↓

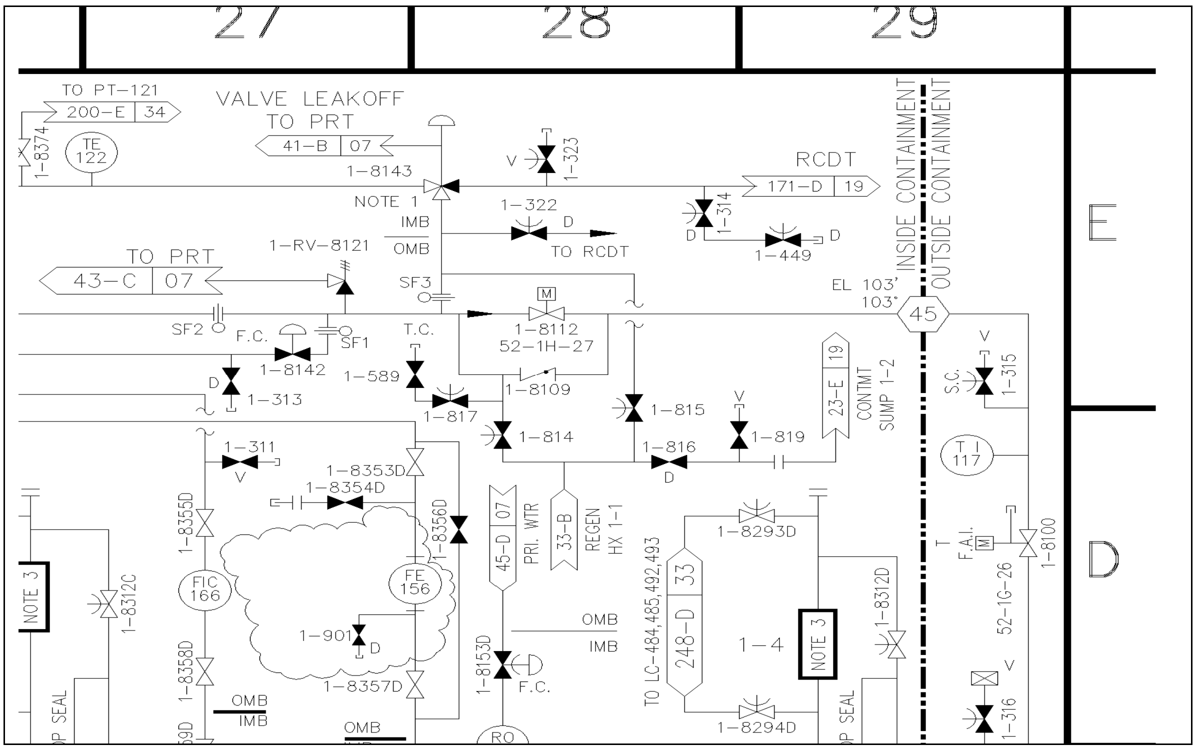
Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

0Examination Outline Cross-Reference	Level	SRO
004 Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Boundary isolation valve leak	**Tier #**	2
	Group #	1
	K/A #	004 A2.03
	Rating	4.2

Question 87 PARENT QUESTION

A leak develops at the inlet of RCP Seal Water Return Stop valve, CVCS-8100.

Which of the following actions will be necessary to stop the leakage and what will be the impact, if any, when a the RCS water balance surveillance to verify RCS operational LEAKAGE is within limits as required by SR 3.4.13.1, is performed after the leakage is isolated?



	To stop the leak CLOSE:	The RCS water balance surveillance:
A.	CVCS-8100.	will NOT be affected.
B.	CVCS-8112	will NOT be affected.
C.	CVCS-8100	will be affected.
D.	CVCS-8112	will be affected.

Proposed Answer: D. CVCS-8112 will be affected.

Explanation:

A. Incorrect. Closing 8100 will not stop the leakage. If the leak was on the outlet, this would be successful. SR 3.4.13.1 will be affected due to the lifting of the relief to the PRT when the leak is isolated by closing 8112.

B. Incorrect. SR 3.4.13.1 will be affected due to the lifting of the relief to the PRT when the leak is isolated by closing 8112.
C. Incorrect. Closing 8100 will not stop the leakage. If the leak was on the outlet, this would be successful. SR 3.4.13.1 will be affected due to the lifting of the relief to the PRT when the leak is isolated by closing 8112.
D. Correct. Closing 8112 will stop the leakage but the relief valve to the PRT will lift. SR 3.4.13.1 will be affected due to the lifting of the relief to the PRT.

Technical References: STP-10C, OVID 106708 sheet 2

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.2	

NOTE: Changed STP-10C to SR 3.4.13.1

Category: S - System Malfunction

Subcategory: 5 - RCS Leakage

Initiating Condition: RCS LEAKAGE for 15 minutes or longer.

EAL:

SU5.1 Unusual Event

RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 minutes.

OR

RCS identified leakage > 25 gpm for ≥ 15 minutes.

OR

Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 minutes. (Note 1)

Note 1: The SM/SEC/ED should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

RCS LEAKAGE - RCS leakage shall be:

a. Identified Leakage

    1. Leakage, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leak off), that is captured and conducted to collection systems or a sump or collecting tank;

    2. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage;

    3. RCS leakage through a steam generator to the secondary system (primary to secondary leakage).

b. Unidentified Leakage

    All leakage (except RCP seal water injection or leak off) that is not identified leakage.

c. Pressure Boundary Leakage

    Leakage (except primary to secondary leakage) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall.

d. RCS leakage outside of the containment that is not considered identified or unidentified leakage per Technical Specifications includes leakage via interfacing systems such as RCS to the Component Cooling Water, or systems that directly see RCS pressure outside containment such as Chemical & Volume Control System, Nuclear Sampling System and Residual Heat Removal system (when in the shutdown cooling mode).

Basis:

ERO Decision Making Information

These conditions apply to leakage into the containment, a secondary-side system (e.g., steam generator tube leakage) or a location outside of containment.

The first and second EAL conditions are focused on a loss of mass from the RCS due to unidentified leakage, pressure boundary leakage or identified leakage (as these leakage types are defined in the plant Technical Specifications) (ref. 2).

The third condition addresses an RCS mass loss caused by an UNISOLABLE leak through an interfacing system (ref. 3, 4, 5).

If the leak is isolated, the RCS barrier was never lost.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. An emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated).

The 15 minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

Escalation of the emergency classification level would be via ICs of Recognition Category R or F.

Below is a summary of classification guidance for steam generator tube leaks:

	Affected SG is FAULTED Outside of Containment?	
Primary-to-Secondary Leak Rate	**Yes**	**No**
Less than or equal to 25 gpm	No classification	No classification
Greater than 25 gpm	Unusual Event per SU5.1	Unusual Event per SU5.1
Requires operation of a standby charging (makeup) pump with letdown isolated (RCS barrier potential loss)	Site Area Emergency per FS1.1	Alert per FA1.1
Requires an automatic or manual ECCS (SI) actuation (RCS barrier loss)	Site Area Emergency per FS1.1	Alert per FA1.1

Background

Manual or computer-based methods of performing an RCS inventory balance are normally used to determine RCS LEAKAGE.

STP R-10C, Reactor Coolant System Water Inventory Balance, is performed to determine the source and flow rate of the leakage (ref. 1).

Escalation of this EAL to the Alert level is via Category F, Fission Product Barrier Degradation, EAL FA1.1.

This IC addresses RCS LEAKAGE which may be a precursor to a more significant event. In this case, RCS LEAKAGE has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	3
G2.3.11 Ability to control radiation releases.	**Group #**	2
	K/A #	G2.3.11
	Rating	4.3

Question 23

According to the bases for ECG 23.3, Containment Ventilation System, the containment vacuum/pressure relief isolation valves:

1. Operation of the valves is limited to a maximum of __________ hours per calendar year and
2. valve travel is limited because the:

A. 1. 90
   2. valves are not qualified to close during an accident

B. 1. 90
   2. containment negative pressure limit could be exceeded if they were fully opened

C. 1. 200
   2. valves are not qualified to close during an accident

D. 1. 200
   2. containment negative pressure limit could be exceeded if they were fully opened

Proposed Answer: C. 1. 200 2. valves are not qualified to close during an accident

Explanation:
SRO Justification, in lieu of flowsheet, per ES-4.2, for 55.43.4, 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 are authorizing releases or emergency exposures.

A. Incorrect. 90 hours is in the bases as a proposed NRC limit. Plausible as the bases states 200 hours is the compromise number reached. Second part is correct.
B. Incorrect. 90 hours is incorrect. the limit is 200 hours. Plausible as this is the number in the bases that was discussed. Second part plausible because there is a negative pressure Tech Spec limit and the valves are opened for containment pressure control, therefore, plausible the valve travel is limited to meet the negative pressure limit.
C. Correct. The limit is 200 hours. The valve travel is limited because they are not qualified to close under accident conditions.
D. Incorrect. First part is correct. The second part is incorrect. Second part plausible because there is a negative pressure Tech Spec limit and the valves are opened for containment pressure control, therefore, plausible the valve travel is limited to meet the negative pressure limit.

Technical References: ECG 23.3

References to be provided to applicants during exam: None

Learning Objective: 66064 - Apply the requirements of System 23 ECGs.

Question Source:	Bank #91 L161	Y
(note changes; attach parent)	Modified Bank #	

	New	
	Past NRC Exam DCPP 10/2016	Y
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.4	

Difficulty: 2.5

Examination Outline Cross-Reference	Level	SRO
029 G2.2.25 – Containment Purge: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	**Tier #**	2
	Group #	2
	K/A #	029 G2.2.25
	Rating	4.2

Question 91 PARENT QUESTION

Per ECG 23.3, Containment Ventilation System, which of the following states the MAXIMUM amount of time the containment purge supply and/or exhaust isolation valves may be open and the reason for the limit?

________________hours to ________________.

A. 200; prevent exceeding the NPDES permit

B. 200; minimize the probability of a LOCA occurring while the valves are open

C. 90; prevent exceeding the NPDES permit

D. 90; minimize the probability of a LOCA occurring while the valves are open

Proposed Answer: B. 200; minimize the probability of a LOCA occurring while the valves are open

Explanation:

A. Incorrect. Plausible because the purge could be considered a discharge (and impact the NPDES), it does not and is not the reason stated in the bases.
B. Correct. The ECG states the valves can be open for 200 hours per calendar year. The bases states the purging time restriction is meant to minimize the probability of a LOCA while conducting purging operations and thereby limit offsite boundary doses.
C. Incorrect. Plausible 90 hours is stated in the Bases as the time the NRC had wanted as a time limit.
D. Incorrect. Plausible 90 hours is stated in the Bases as the time the NRC had wanted as a time limit.

Technical References: ECG 23.3
References to be provided to applicants during exam: none
Learning Objective: 66064 -Apply the requirements of System 23 ECGs.

Question Source:	Bank # 98 DCPP NRC Exam L091, 07/2011	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Past NRC Exam	Yes
	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.2	

The SRO must know the bases for the limitation of the ECG (which is a license document).

Difficulty: 2.5

Rev 1 – Added "MAXIMUM" to stem

23.0 HEATING, VENTILATION AND AIR CONDITIONING

23.3 Containment Ventilation System

ECG 23.3 Operation with the vacuum/pressure relief isolation valves open up to 50° shall be limited to less than or equal to 200 hours during a calendar year.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The vacuum/pressure relief isolation valves open up to 50° for more than 200 hours during a calendar year. OR The vacuum/pressure relief isolation valves open beyond 50°.	A.1 Close the open isolation valves. OR A.2 Isolate the penetration(s) flowpath(s).	1 hour 1 hour
B. Required action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 23.3.1	Determine the cumulative time that the vacuum/pressure relief isolation valves have been open during a calendar year.	7 days

BASES

BACKGROUND	ECG 23.3 was developed to relocate the 200 hour per year time restriction on operation of containment ventilation flow paths from TS 3.6.1.7 and 4.6.1.7.2 to an ECG (ECG 23.3). In accordance with TS 3.6.3, each 48 inch containment purge valve is sealed closed during Modes 1 to 4. Similarly, each 12 inch vacuum/pressure relief valve is also closed except when these valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the vacuum/pressure relief valves to be open. Refer to TS Bases Section 3.6.3 for details (Reference 9).
APPLICABLE SAFETY ANALYSES	Use of the containment purge lines is restricted to the vacuum/pressure relief line to ensure that the site boundary dose guidelines of 10 CFR 50.67(b)(2) and Section 4.4, Table 6 of RG 1.183 would not be exceeded in the event of a LOCA during containment purging operations. The vacuum/pressure relief valves must be blocked to open no more than 50° because these valves have not been qualified to close under accident conditions (References 7, 8, and 9).
LCO	The purging time restriction is meant to minimize the probability of a LOCA while conducting purging operations and thereby limit offsite boundary doses. The 200 hour per calendar year restriction reflects a regulatory compromise between an NRC proposed limit of 90 hours per year and a PG&E counter-proposal of 1000 hours per year (References 3, 5).
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES (reference 6). Therefore, the time limitations on containment purging operations only applies in MODES 1, 2, 3, and 4.

(continued)

BASES (continued)

ACTIONS	A.1 and A.2 In the event the LCO is not satisfied, the open isolation valves must be closed, or the penetration flowpaths isolated, within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident occurring during periods when the vacuum/pressure relief isolation valves are open is minimal (Reference 6). B.1 and B.2 If the vacuum/pressure relief isolation valves cannot be closed within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems (Reference 6).
SURVEILLANCE REQUIREMENTS	SR 23.3.1 Operation with the vacuum/pressure relief isolation valves open up to 50° is limited to 200 hours during a calendar year. The 200-hour limit became effective after initial criticality. The total time the vacuum/pressure relief isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience (References 7, 8, and 9).
REFERENCES	1. License Amendment 135/135 2. CTS Bases 3/4.6.1.7 3. DCM S-23A 4. DCL-88-176 5. DCM T-16 6. ITS Bases 3.6.2 7. DCL-15-069 8. License Amendment 230/232, dated April 27, 2017 9. TS Bases Section 3.6.3

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	3
G2.3.12 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, or alignment of filters.	**Group #**	3
	K/A #	G2.3.12
	Rating	3.7

Question 24

Unit 1 is in MODE 1. Plant and radiological conditions are stable.

Operators are preparing to make multiple containment entries per RCP D-230, Radiological Control for Containment Entry.

Which of the following action(s) listed below is/are the responsibility of the Unit 1 Shift Foreman?

1. Authorizing the containment entry
2. Maintaining possession of the MIDS keys
3. Conducting the Containment Pre-Entry Brief

A. 1 only

B. 2 only

C. 1 and 3

D. 2 and 3

Proposed Answer: A. 1. only

Explanation:
SRO Justification, in lieu of flowsheet, per ES-4.2, for 55.43.4, 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 are authorizing releases or emergency exposures. Containment entry meets this criteria also.

A. Correct. Per RCP D-230, SFM authorization must be obtained. (attachment 10.1)
B. Incorrect. MIDS keys are kept by RP. Plausible because the Shift Foreman maintains control over a number of keys for the unit.
C. Incorrect. First part is correct. Second part incorrect. Plausible as the tailboard is done by RP for routine entries but conducted by the SFM for rapid containment entry.
D. Incorrect. MIDS keys are kept the RP. Plausible because the SFM maintains possession of most keys. The Containment Pre-Entry Tailboard is conducted by the SM as part of a rapid containment entry in OP AP-31. In RCP-230, RP is responsible for assuring completion of the Containment Entry Checklist, which includes typical tailboard/pre-job

Technical References: RCP D-230 attachment 1

References to be provided to applicants during exam: None

Learning Objective: 41660 - Describe the general authorities and responsibilities of the Unit Shift Foreman during normal and abnormal plant operations

Question Source:	Bank #98 L161	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 10/2016	Y
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.4	

Difficulty: 2.5

Examination Outline Cross-Reference	Level	SRO
G2.3.12 - Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc	**Tier #**	3
	Group #	3
	K/A #	G2.3.12
	Rating	3.7

Question 98 PARENT QUESTION

Unit 1 is in MODE 1. Plant and radiological conditions are stable.

Operators are preparing to make multiple containment entries per RCP D-230, Radiological Control for Containment Entry.

Which of the following action(s) listed below is/are the responsibility of the Shift Foreman?

1. Authorizing the containment entry
2. Maintaining possession of the MIDS keys
3. Conducting the Containment Pre-Entry Tailboard

A. 1 only

B. 1 and 2

C. 2 and 3

D. 1 and 2 and 3

Proposed Answer: A. 1 only

Explanation:
Per RCP D-230, SFM authorization must be obtained.
A. Correct. The SFM authorizes the entry (attachment 10.1).
B. Incorrect.MIDS keys are kept by RP
C. Incorrect. MIDS keys are kept the RP. Plausible because the SFM maintains possession of most keys. The Containment Pre-Entry Tailboard is conducted by the SM as part of a rapid containment entry in OP AP-31. In RCP-230, RP is responsible for assuring completion of the Containment Entry Checklist, which includes typical tailboard/pre-job
D. Incorrect. MIDS keys are kept the RP. Plausible because the SFM maintains possession of most keys. The Containment Pre-Entry Tailboard is conducted by the SM as part of a rapid containment entry in OP AP-31. In RCP-230, RP is responsible for assuring completion of the Containment Entry Checklist, which includes typical tailboard/pre-job.

Technical References: RCP D-230
References to be provided to applicants during exam: none
Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Past NRC Exam	No
	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.4	

SRO responsibilities for containment entry.

Difficulty: 2.5
Rev 1 – rewrote question

4. PREREQUISITES

4.1 Ensure personnel entering containment comply with the applicable RWP.

5. PRECAUTIONS AND LIMITATIONS

5.1 Non-emergency entries shall be made in accordance with RCP D-230, "Radiological Control for Containment Entry."

5.2 During time critical actions where Shift Manager requires an emergency containment entry, the following are NO ENTRY locations:[Ref. 7.2.2]

- 91' bioshield (dose rates at power may easily exceed admin and/or regulatory limits)
- Posted VHRA
- Reactor cavity
- Areas greater than 800 mr/hr, unless briefed on anticipated dose rate alarm

2. DISCUSSION

2.1 During MODES 1 through 4, access to containment is sometimes required for inspections, surveillance tests, and maintenance. There are special considerations associated with radiological and non-radiological hazards for entries at power. This procedure focuses on expected recurring issues.

2.2 OP I-1:I is used in conjunction with this procedure. OP I-1:I provides instruction for entering and exiting containment including required notifications, airlock operation, energizing/de-energizing containment lighting, and verification of movable incore detector system (MIDS).[Ref 9.4]

2.3 There is a leak rate test panel outside the airlock door that provides indication of an in progress airlock test.

2.4 STP M-45B is used by the entrants in conjunction with this procedure to maintain containment cleanliness.

2.5 Respiratory protection is evaluated and implemented per RCP D-200 for radiological use and OM6.ID10 for industrial safety.

2.6 Checklist data is recorded weekly when RCP D-800 samples are collected.

2.6.1 The current data is recorded prior to entry within 24 hours.

3. DEFINITIONS

3.1 Non-Radiological Hazards: Hazards that include but are not limited to oxygen deficiency, explosive or toxic gases, heat stress, steam leakage, standing oil or water, poor lighting, and noise.

3.2 Radiological Hazards: Hazards that include airborne particulates, radioiodines, noble gases, tritium; gamma radiation, neutron radiation, and surface contamination.

4. RESPONSIBILITIES

4.1 RP foreman or designee is responsible for ensuring completion of Attachment 1, Form 69-20560, "Containment Entry Checklist."

4.2 Personnel entering containment are responsible for obtaining and completing the STP M-45 data sheet.

4.3 DCPP safety is responsible for overall safety requirements including but not limited to:

4.3.1 Establishing specific safety precautions as necessary for the entry.

4.3.2 Determining atmospheric sample requirements for non-radiological hazards (e.g., oxygen and LEL) outside routine sampling included in this procedure.

7. INSTRUCTIONS

7.1 Checklist

NOTE: OP I-1:I provides instruction for airlock operation and containment lighting.

7.1.1 Initiate or obtain an initiated Attachment 1, Form 69-20560, "Containment Entry Checklist."

a. Same checklist may be used for one day.

b. Ensure parts 1 and 2 of the checklist are complete.

7.1.2 Brief all entrants for each entry using the checklist.

7.2 Radiological Airborne Considerations

NOTE: RE-11/12 air samples to support containment entry are typically taken weekly.[Ref 9.7]

7.2.1 Check RE-11/12 are in service before recording data from the PDN server. (PDN displays data when monitor OOS.)

7.2.2 Compare weekly and current readings.

a. IF readings are outside of the range listed below,
THEN prior to entry, ensure new samples are taken.
- RE-11 readings ± 500 cpm.
- RE-12 readings ± 100 cpm.

7.2.3 IF RE-11/12 is out of service,
THEN ensure samples are taken within 24 hours prior to entry,
OR consider the last sample valid if both of the following conditions are met:

a. CTMT structure sump level has been stable for the last 24 hours.

b. RE-2 is in service and the requirements of step 7.4.4 are met.

7.2.4 Based on airborne results, request operations ventilate containment or initiate iodine removal as determined by RP supervisor.

Containment Entry Checklist

Part 1: Weekly Data Unit: ____________ Rx Power % ____________

Sample Date/Time: ______________________/______________ Survey #: ___________

DAC Fraction
Particulate: ________________
Iodine: ____________________
Noble Gas: ________________
Tritium: __________________
Total: ____________________

RE-11: ________________cpm
RE-12: ________________cpm
RE-2: _________________mr/h

Oxygen: ____________%
Allowable 19.5 % to 23.5 %

LEL: _______________%
Allowable ≤ 20 %

Part 2: Current Data Sample Date/Time: ______________/_________ Rx Power ________ %

Oxygen _____________% Allowable 19.5% to 23.5% LEL ___________% Allowable ≤ 20 %
RE-11: _____________ cpm *Allowable ± 500 cpm RE-12: _____________ cpm *Allowable ± 100 cpm
RE-2: ___________ mr/hr *Allowable ± 10 mR/hr *Compared to weekly results

Job Description: ____Pager # ___________

Part 3: Approvals and Checks

1.	SFM _____________________________________ has authorized entry?		☐ Yes ☐ No
2.	Anticipated alarms during the entry?		☐ No ☐ Yes
3.	Anticipated Rx power changes? RPM and operations manager or director permission required for entry during power changes.		☐ No ☐ Yes
4.	Containment pressure: __________psig. Is pressure 0 ±1 psig?		☐ Yes ☐ No
5.	MIDS stored or in a known approved location? (Refer to OP I-1:I)		☐ Yes ☐ No
6.	RP has possession of MIDS keys?		☐ Yes ☐ No
7.	RE 11/12 weekly and current readings acceptable?	☐ OOS	☐ Yes ☐ No
8.	RE-2 weekly and current readings acceptable?	☐ OOS	☐ Yes ☐ No
9.	Oxygen, LEL, and any other specified industrial sample results acceptable?		☐ Yes ☐ No
	a. Portable atmospheric monitoring required (e.g., Altairs) per step 5.2?		☐ No ☐ Yes
	b. Safety or RPM notified of increased LEL or other increased results?	☐ N/A	☐ Yes ☐ No
10.	Review job scope, location, component, radiological conditions.		☐ Done
	a. Neutron time keeping required?		☐ No ☐ Yes
	b. RWP correct for job scope? RWP #: _______		☐ Yes ☐ No
	c. Review RWP instructions including dose/dose rate settings. Dose: _________Dose Rate: ____________		☐ Done
	d. Discuss high noise and ability to hear PED alarm.		☐ Done
	e. Discuss when to use 3-way communication.		☐ Done
	f. All briefing requirements for HRA, LHRA, VHRA complete?		☐ Yes ☐ No
	g. All required paperwork for VHRA and LHRA complete?	☐ N/A	☐ Yes ☐ No
11.	Review expected individual dose for this entry.		☐ Done
12.	Anticipated PED alarms?		☐ No ☐ Yes
13.	Any radiological hold points?		☐ No ☐ Yes
14.	Review safety issues such as oil hazards on walkways, temperature, lighting, PPE, high noise, two entrant rule, fall protection, etc. OP I-1:I provides lighting instructions.		☐ Done
15.	Review any applicable OE.		☐ Done
16.	Review Control Room and RP notification requirements. (Refer to OP I-1:I)		☐ Done

Completed By: _________________________ LAN ID:_______ Date/Time: __________________

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	3
G2.4.18 Knowledge of the specific bases for emergency and abnormal operating procedures	**Group #**	4
	K/A #	G2.4.18
	Rating	4.0

Question 25

1. Manual action(s) with a specified completion time limit to meet a plant licensing basis requirement at DCPP are identified as:
2 One such licensing document that contains events mitigated by these actions is: ________.

A. 1. Time Critical Operator Actions (TCOA)
   2. FLEX

B. 1. Time Critical Operator Actions (TCOA)
   2. the FSAR

C. 1. Time Sensitive Operator Actions (TSOA)
   2. FLEX

D. 1. Time Sensitive Operator Actions (TSOA)
   2. the FSAR

Proposed Answer: B. 1. Time Critical Operator Actions (TCOA)
2. the FSAR

Explanation:

SRO Justification (in lieu of worksheet) from ES-4.2 - 3. Examination authors should use the 10 CFR 55.43(b) topic-based guidance and examples (a–g below) when developing SRO-level questions. This question meets the criteria for 55.43.1

a. Conditions and Limitations in the Facility License [10 CFR 55.43(b)(1)]Examples of SRO-only examination items for this topic include the following:

UFSAR and licensing documents are SRO knowledge

A. Incorrect. First part is correct. Second part is a risk based program and forms the bases for actions identified as TSOA. (Sources of time sensitive operator actions (TSOAs) include EDMG, FLEX, Probabilistic Risk Assessment (PRA) and risk-based analyses or programs such as NFPA 805.) Plausible if it is thought because there are times associated with them, that these are TCOAs.
B. Correct. DCPP's licensing basis addresses automatic and manual actions for accident mitigation. Such actions may be in response to a fire event, station blackout, Steam Generator Tube Rupture (SGTR) or other event in the licensing basis. In some cases, credit is taken in the plant licensing basis for manual actions that are performed within a specified time; these actions may be described as Time Critical Actions (TCOAs). (Tech Specs, Licensing commitments, UFSAR – referred to as “FSAR”)
C. Incorrect. Both parts are incorrect. Plausible because TSOAs are also tracked but used for risk based programs. Second part incorrect but plausible as there are timely actions associated with FLEX procedures.
D. Incorrect. First part incorrect, TSOAs are tracked but are associated with Risk based programs. Second part is correct.

Technical References: OP1.ID2

References to be provided to applicants during exam: None

Learning Objective: 77253 From memory, describe the guidance for time critical operator actions per OP1.ID2 and OP1.DC10.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	N
Question History:	Last Two NRC Exams	N
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.1	

Difficulty: 2

1.1.5 Verify that any changes to the plant or to procedures or protocols do not invalidate credited action times. TCOAs are identified in a variety of sources, which include:

a. Technical Specifications

b. Station Blackout Analysis

c. Licensing Commitments

d. Fire Events[Ref 7.7]

e. Design engineering documents

f. EOPs and AOPs

g. Administrative operating procedures such as OP1.DC10, which affects conduct of operations

h. Updated Final Safety Analysis Report (UFSAR)

1.1.6 Sources of time sensitive operator actions (TSOAs) include EDMG, FLEX, Probabilistic Risk Assessment (PRA) and risk-based analyses or programs such as NFPA 805.

1.1.7 Evaluate and document time sensitive operator actions.

1.2 The program directive CF1, "Configuration Management," controls the identification of new or modified TCOAs due to engineering processes.

1.3 This program augments the procedure validation process of AD1.DC12.

1.4 This program specifically excludes security related TCOAs. Security related TCOAs are Safeguards Information under the control of the plant security program, which already identifies, tracks and validates these actions.

1.5 Validation of TCOAs is not considered part of operator training programs. TCOAs are validated using the Simulator and JPMs. This procedure controls any type of TCOA validation. TQ1, "Personnel Training and Qualification," and TQ2, "Accredited Training Programs" control the performance evaluation of the simulator crew or JPM performer.

2. DISCUSSION

2.1 DCPP's licensing basis addresses automatic and manual actions for accident mitigation. Such actions may be in response to a fire event, station blackout, Steam Generator Tube Rupture (SGTR) or other event in the licensing basis. In some cases, credit is taken in the plant licensing basis for manual actions that are performed within a specified time; these actions may be described as Time Critical Actions (TCOAs).

2.2 Typically, plant operations group performs the majority of time critical actions, with relatively few time critical actions performed by other plant personnel. Therefore, this standard focuses on plant operations.

2.3 Validation provides documented confirmation that the specified TCOA is achievable. Periodic revalidation demonstrates the continued ability to meet TCOAs. Periodic revalidation is a valuable tool for detecting an unexpected challenge to TCOA completion time, which may occur due to the aggregate of procedure and protocol changes and equipment modifications over time.

2.4 ANS 58.8-1994, "Time Response Design Criteria for Safety Related Operator Actions" establishes timing requirements used in the design of safety related systems for nuclear power plants. Design engineers use these criteria to determine whether safety-related systems can be initiated by operator action or require automatic initiation. It is beyond the model and data base of ANS 58.8-1994 to use its timing requirements to calculate actual operator action times.

2.5 Credit for meeting TCOA validation times may be taken during scheduled Simulator or JPM training. Due to the different purposes of this program and the training program, failure to meet a TCOA under this program does not constitute a failure under the training program. TCOAs provide confidence of the operators' ability to perform required actions to meet licensing commitments. The simulator is not used to determine the engineering basis for a TCOA since it is not an engineering credited analysis tool. The simulator is for training and can be used to prove TCOA times are met based on the operator's action and moving through procedures, but it is not used to determine those times.

3. DEFINITIONS

3.1 Engineering Document: Any document prepared by the engineering organization, or any vendor document reviewed and approved by engineering for use.

3.2 Event Limit: The earliest time at which a limiting design function would be exceeded if a safety related function has not been completed. See ANS 58.8-1994.

3.3 Minimum Shift Staffing: The minimum number of personnel required per shift as defined in the plant administrative procedures.

3.4 Plant Licensing Basis: Documents used to design and operate the plant within set parameters determined by the NRC. Events mitigated by TCOAs are contained within the UFSAR, NRC Safety Evaluation Report (SER) commitments, or other design or licensing basis documents and supporting documents to ensure nuclear safety.

3.5 Recovery Action Feasibility Assessment (RAFA): Is NFPA 805 TSOA that replaced Appendix R TCOA fire actions. The RAFA contains all validation information and will not be added to Attachment 2 TSOA spread sheet. It is located at PG&E Calculation SAP No. 9000041468 (Legacy No. M-1179), Recovery Action Feasibility Assessment.

3.6 Revalidation: Periodic verification of the continued ability of operators to perform TCOAs within the required time. Revalidation refers to the validation process used for revised or modified (process changed) TCOAs and the periodic evaluation of unchanged TCOAs.

3.7 Rim Pull: Amount of force that must be applied to the handwheel rim to open or close a valve.

3.8 Simulator Validation: Execution of procedures using the plant-specific simulator during a simulated event.

3.9 Time Critical Operator Action (TCOA): A time-constrained manual action or series of actions which are credited in the safety analyses as part of the primary success path for mitigating design basis accidents. Manual operator actions required to support the credited strategy in the safety analysis (both implicitly and explicitly) are also TCOA.

3.10 Time Critical/Sensitive Operator Action Database: Operations maintained documentation in D2 (DCPP >Operations>Time Critical Operator Action>TCOA).

3.11 Time Sensitive Operator Action (TSOA): A time-constrained manual action or series of actions that are not credited in the plant Design Basis but must be completed within a specified time to meet requirements in programs, procedures, and documents outside the plant's licensing basis.

3.12 Validation: Performance of a new TCOA on the simulator or by in-plant walkthrough, or both, to verify the operators can perform the TCOA within the required time using the applicable procedures, including all required human performance protocols.

3.13 Walkthrough Validation: A step-by-step in-plant walkthrough of procedures by plant personnel, simulating manipulation of controls and equipment.