

U.S. Nuclear Regulatory Commission
Diablo Canyon RO Written Examination

Applicant Information

Name: **KEY**

Date: 9 February, 2018

Facility/Unit: Diablo Canyon

Region: I II III IV

Reactor Type: W CE BW GE

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. Examination papers will be collected 6 hours after the examination begins

Applicant Certification

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

Applicant's Signature

Results

Examination Value _____ Points

Applicant's Score _____ Points

Applicant's Grade _____ Percent

U.S. Nuclear Regulatory Commission
Diablo Canyon SRO Written Examination

Applicant Information

Name: **KEY**

Date: 9 February, 2018

Facility/Unit: Diablo Canyon

Region: I II III IV

Reactor Type: W CE BW GE

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 SRO-only items. You have 9 hours to complete the combined examination.

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 Values _____ / _____ / _____ Points

Applicant's Scores _____ / _____ / _____ Points

Applicant's Grade _____ / _____ / _____ Percent

Multiple Choice (Circle your choice)

NAME: _____ KEY _____

If you change your answer, write your selection in the blank and initial.

- 1. A B C D
- 2. A B C D
- 3. A B C D
- 4. A B C D
- 5. A B C D
- 6. A B C D
- 7. A B C D
- 8. A B C D
- 9. A B C D
- 10. A B C D
- 11. A B C D
- 12. A B C D
- 13. A B C D
- 14. A B C D
- 15. A B C D
- 16. A B C D
- 17. A B C D
- 18. A B C D
- 19. A B C D
- 20. A B C D
- 21. A B C D
- 22. A B C D
- 23. A B C D
- 24. A B C D
- 25. A B C D
- 26. A B C D
- 27. A B C D
- 28. A B C D
- 29. A B C D
- 30. A B C D
- 31. A B C D
- 32. A B C D
- 33. A B C D
- 34. A B C D
- 35. A B C D
- 36. A B C D
- 37. A B C D
- 38. A B C D
- 39. A B C D
- 40. A B C D
- 41. A B C D
- 42. A B C D
- 43. A B C D
- 44. A B C D
- 45. A B C D
- 46. A B C D
- 47. A B C D
- 48. A B C D
- 49. A B C D
- 50. A B C D

Multiple Choice (Circle your choice)

NAME: KEY

If you change your answer, write your selection in the blank and initial.

*Question deleted. See Exam Report 2018-301 for details.

- 51. A B C D
- 52. A B C D
- 53. A B C D
- 54. A B C D
- 55. A B C D
- 56. A B C D
- 57. A B C D
- 58. A B C D
- 59. A B C D
- 60. A B C D
- 61. A B C D
- 62. A B C D
- 63. A B C D
- 64. A B C D
- 65. A B C D
- 66. A B C D
- 67. A B C D
- 68. A B C D
- 69. A B C D
- 70. A B C D
- 71. A B C D
- 72. A B C D
- 73. A B C D
- 74. A B C D
- 75. A B C D

- 76*. A B C D
- 77. A B C D
- 78. A B C D
- 79. A B C D
- 80. A B C D
- 81. A B C D
- 82. A B C D
- 83. A B C D
- 84. A B C D
- 85. A B C D
- 86. A B C D
- 87. A B C D
- 88. A B C D
- 89. A B C D
- 90. A B C D
- 91. A B C D
- 92. A B C D
- 93. A B C D
- 94. A B C D
- 95. A B C D
- 96. A B C D
- 97. A B C D
- 98. A B C D
- 99. A B C D
- 100. A B C D

Examination Outline Cross-Reference
003 K4.04 Knowledge of RCPS design feature(s) and/or interlock(s) which provide for the following: Adequate cooling of RCP motor and seals

Level	RO
Tier #	2
Group #	1
K/A #	003 K4.04
Rating	2.8

Question 01

A steam break occurs inside containment on Unit 1.

Containment pressure peaks at 15 psig.

- 1) Cooling to the RCP #1 Seals has been:
- 2) Cooling to the RCP bearing oil coolers has been:
 - A. 1) maintained throughout the event
2) maintained throughout the event
 - B. 1) isolated
2) maintained throughout the event
 - C. 1) isolated
2) isolated
 - D. 1) maintained throughout the event
2) isolated

Proposed Answer: A. 1) maintained throughout the event
2) maintained throughout the event

Explanation:

- A. Correct. Phase A isolates seal return from the #1 RCP seals to the VCT and directs it to the PRT, however, as long as FCV-128 is still open (is not closed by phase A), backpressure will continue to supply charging to the seals. NOTE, in the EOPs, the crew is instructed to "maintain 8 to 13 gpm to RCP seals". Bearing oil coolers lose CCW cooling if Phase B actuates, (22 psig) Note, if Phase B actuates, then cooling would be lost to both the thermal barrier and the oil coolers..
- B. Incorrect. Plausible if its assumed that the seal cooling is lost when phase A actuates (containment pressure of 3 psig), this does stop charging injection, thru FCV-128 via the normal flowpath however the thermal barrier is still cooled by CCW Cooling to the motor oil coolers is maintained because containment pressure remained less than the Phase B setpoint.
- C. Incorrect. Plausible if either its assumed that containment pressure rose above the Phase B setpoint or its thought both are lost at the Phase A setpoint of 3 psig.
- D. Incorrect. Plausible – cooling is maintained to the #1 seal and its thought that motor oil coolers lose cooling at Phase A or Phase B setpoint was exceeded but that cooling is maintained to the #1 seal solely by RCS flowing past the seal.

Technical References: LA6

References to be provided to applicants during exam: None

Learning Objective:State the purpose of RCP components. (35737)

Question Source: Bank #

(note changes; attach parent)	Modified Bank #	
Question History:	New	X
Question Cognitive Level:	Last Two NRC Exams	No
	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	
Difficulty: 2.0		

L162 Answer Key

raising pressure will force the gas back the VCT

- D. Incorrect. Normal gas in the VCT is hydrogen which is potentially explosive, nitrogen is not. If it assumed nitrogen is also explosive, this is feasible. Venting is the method of removal..

Technical References: OP A-2:I.

References to be provided to applicants during exam: None

Learning Objective: 5093 -Discuss significant precautions and limitations associated with the CVCS

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content: Difficulty: 3.0	55.41.5	

L162 Answer Key

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	1
K/A #	005 G2.4.4
Rating	4.5

005 G2.4.4 RHR: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Question 03

Following a design basis LOCA, loss of which of the following would meet the entry conditions for EOP ECA-1.1, Loss of Emergency Coolant Recirculation?

Loss of:

- A. both RHR pumps
- B. both Charging pumps and a RHR pump
- C. both SI pumps and both Charging pumps
- D. one train of ECCS pumps (RHR, SI and ECCS CCP)

Proposed Answer: A. both RHR pumps

Explanation:

Note: RO and SRO objective to know major actions and entry conditions for EOPs. Question is a simple entry condition question - appropriate for RO (on SRO flowchart, "Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?" would be answered "yes" or RO knowledge.

- A. Correct. ECA-1.1 is entered when the plant cannot establish at least one train of recirculation with an RHR pump and the containment sump. Without an RHR pump, the SI or charging pumps would not have a suction source.
- B. Incorrect. Loss of the SI pumps would degrade recirculation but as long as there is an RHR pump, there is a recirculation flowpath.
- C. Incorrect. Charging pumps are the major contributors to degrading conditions leading to inadequate core cooling if there is a small break but an absence of charging pumps does not meet the entry conditions for ECA-1.1, only the lack of an RHR pump.
- D. Incorrect. RHR pump (or lack of a flowpath) is the only condition that causes entry into ECA-1.1. In E-1.3 step 12, part of the step is to check for NO flow from any of the flow indicators, RHR, SI, charging and if there is no flow on all of them (due to the first part, no RHR pump), then the RNO is go to ECA-1.1. If this step is remembered incorrectly, this would seem to be the answer.

Technical References: ECA-1.1 section 2, Symptoms or Entry Conditions, E-1.3 step 12, LPE1C

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 Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X

10CFR Part 55 Content:

Comprehensive/Analysis
55.41.10

Difficulty: 2.0

L162 Answer Key

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	1
K/A #	005 K1.10
Rating	3.2

005 K1.10 -Knowledge of the physical connections and/or cause effect relationships between the RHRS and the following systems: Containment Spray System (CSS)

Question 04

Which of the following valves, when CLOSED, provides a direct input for completing the interlock to OPEN the RHR to Containment Spray Isolation valve, 9003A.

- A. 9001A, Containment Spray Pump 11 Discharge
- B. 8701, RHR Suction
- C. 8809A, RHR to Cold Legs 1 and 2
- D. 8804A, RHR Hx 12 Discharge to CHG and SI pumps

Proposed Answer: B. 8701, RHR Suction

Explanation:

- A. Incorrect. The pumps are stopped and the pump discharge valves are closed when aligning CS from RHR. However, it is not interlocked with the operation of 9003 A/B. Plausible to believe the normal suction valve(s) must be closed to open the flowpath from RHR.
- B. Correct. In order to open 9003 A or B, the associated containment sump suction valve, 8982 A or B must be open and EITHER 8701 or 8702 must be closed. Closing 8701 provides an input to the interlock to open the valve.
- C. Incorrect. 8809A does not impact the operation of the valve, but is operated (closed) just prior to opening 9003A in E-1.3.
- D. Incorrect. 8804A (and 8804B) are opened to provide suction to charging and SI pumps. Because RHR is being aligned to Cont Spray, plausible that 8804A must be closed to ensure an adequate suction path to the containment spray pumps.

Technical References: LB-2, page 28, PEP V-7B

References to be provided to applicants during exam: None

Learning Objective: Analyze automatic features and interlocks associated with the RHR System. (35317)

Question Source:

(note changes; attach parent)

Bank #	
Modified Bank	
New	X
Previous NRC Exam	No
Last Two NRC Exams	No
Memory/Fundamental	X
Comprehensive/Analysis	
55.41.7	

Question History:**Question Cognitive Level:****10CFR Part 55 Content:**

Difficulty: 2.3

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	1
K/A #	006 A4.11
Rating	4.2

006 A4.11 Ability to manually operate and/or monitor in the control room: Overpressure protection system. (CFR: 41.7 / 45.5 to 45.8).

Question 05

GIVEN:

- RCS Cold Leg temperatures are 270°F
- RCS pressure is 365 psig
- LTOP is in service

An RCS transient occurs and RCS pressure begins rising at 5 psig/minute.

Assuming the pressure rise continues at 5 psig/minute, 1 PORVs will initially open in approximately 2 minutes.

- A. 1) two 2) five
- B. 1) three 2) five
- C. 1) two 2) fourteen
- D. 1) three 2) fourteen

Proposed Answer: C. 1) two 2) fourteen

Explanation:

- A. Incorrect. Only two PORVs open, however, in five minutes, pressure will be at 390 psig, LTOP actuates at 435 psig. 390 psig is plausible because above 390 psig, RCS to RHR valves 8701 and 8702 are prevented from opening.
- B. Incorrect. Only two PORVs open. PCV-474 is not safety related and not part of the LTOP circuit. In five minutes, pressure will be at 390 psig, LTOP actuates at 435 psig. 390 psig is plausible because above 390 psig, valves 8701 and 8702, RCS to RHR valves are prevented from opening.
- C. Correct. PORVs PCV-455C and 456 are operated by LTOP. In fourteen minutes, pressure will be 435 psig which is the lift setpoint.
- D. Incorrect. Time to open is correct, however, only two of the PORVs open.

Technical References: OIM A-1-2 and A-4-7, PTLR

References to be provided to applicants during exam: none

Learning Objective: 36923 - Analyze automatic features and interlocks associated with the Pzr, Pzr Pressure and Level Control System

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Past NRC Exam

No

Question History:

Last Two NRC Exams

No

Question Cognitive Level:

Memory/Fundamental

Comprehensive/Analysis

X

10CFR Part 55 Content:

55.43.7

Difficulty: 2.7

Examination Outline Cross-Reference

007 A3.01 Ability to monitor automatic operation of the PRTS, including: Components which discharge to the PRT

Level	RO
Tier #	2
Group #	1
K/A #	007 A3.01
Rating	2.7

Question 06

Unit 1 is at 100% power. PRT level and pressure are at normal operating level and pressure.

A pressure transient causes letdown pressure to begin to rise.

- 1) Letdown Relief valve, RV-8117 will lift at a setpoint of _____ psig.
 - 2) Assume when RV-8117 lifts, PRT pressure begins to rise at 5 psig/minute. The PRT rupture disks would rupture in approximately _____ minutes.
- A. 1) 450 2) 3
- B. 1) 600 2) 3
- C. 1) 450 2) 19
- D. 1) 600 2) 19

Proposed Answer: D. 1) 600 2) 19

Explanation:

The auto actions to meet the KA are the lifting of the relief valve at 600 psig, which discharges to the PRT and the pressure at which the rupture disc fails of 100 psig. Normal level in the PRT is approximately 85%. Normal pressure is approximately 3 psig.

A. Incorrect. Closing of 8152 will cause relief valve 8117 at 600 psig. 450 psig is the setpoint for the RHR to PRT relief valve. In 3 minutes pressure is only about 20 psig. Plausible if the 85% level is confused as 85 psig.

B. Incorrect. Closing of 8152 will cause relief valve 8117 at 600 psig. In 3 minutes pressure is only about 20 psig. Plausible if the 85% level is confused as 85 psig..

C. Incorrect. Closing of 8152 will cause relief valve 8117 at 600 psig. 450 psig is the setpoint for the RHR to PRT relief valve.

D. Correct. Closing of 8152 will cause relief valve 8117 at 600 psig. The rupture disc setpoint is 100 psig. Normal pressure is approximately 3 psig, so it would take about 19 minutes for pressure to rise to the setpoint

Technical References: LA-4B, LB-2, LB-1A, OP B-1A:XII

References to be provided to applicants during exam: None

Learning Objective: 40573 Describe PRT system components

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Question History:

Last Two NRC Exams

Question Cognitive Level:

Memory/Fundamental

Comprehensive/Analysis

X

10CFR Part 55 Content:

55.41.7

Difficulty: 2.3

Examination Outline Cross-Reference

008 K3.01 - Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: Loads cooled by CCWS

Level	RO
Tier #	2
Group #	1
K/A #	008 K3.01
Rating	3.4

Question 07

Unit 1 is at 100% power, at the middle of core life.

Instrument air is lost to TCV-130, Letdown Heat Exchanger CCW outlet valve.

Which of the following describes the effect on letdown system temperature and the potential reactivity effect?

Letdown temperature ____1)____ and reactor power ____2)____.

- A. 1) rises 2) rises
- B. 1) rises 2) lowers
- C. 1) lowers 2) lowers
- D. 1) lowers 2) rises

Proposed Answer: D. 1) lowers 2) rises

Explanation:

- A. Incorrect CCW valve TCV-130 fails open on of instrument air. Failing open causes CCW flow through the heat exchanger to increase and letdown temperature to decrease (not rise) Plausible – power will rise.
- B. Incorrect. Plausible because this would be correct if TCV-130 failed closed.
- C. Incorrect. Letdown temperature will lower but the effect is to remove boron. Boron is released and power rises. Plausible if the effect of lowering letdown temperature is reversed (boron is released – lowering power)
- D. Correct. TCV-130 fails open. This will lower letdown temperature. At full power, MTC is negative, lowering letdown temperature removes boron from letdown, which is a positive reactivity effect and power will rise.

Technical References: LF-2, LPA-9

References to be provided to applicants during exam: None

Learning Objective: State which way air operated valves fail on loss of air. (9842)

Question Source:	Bank #8 DCPD NRC 04/2016	X
(note changes; attach parent)	Modified Bank #	
	New	
	Previous NRC Exam	Yes
Question History:	Last Two NRC Exams	Yes
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content: 55.41.7

Difficulty: 2.3

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	1
K/A #	008 K2.02
Rating	3.0

**008 K2.02 Knowledge of bus power supplies to the following:
CCW pump, including emergency backup.**

Question 08

Unit 2 Emergency Diesel Generators (EDG) are supplying their respective 4 kV Vital AC buses.

Which EDGs are supplying power to CCW pump 22 and CCW pump 23?

	<u>CCW pump 22</u>	<u>CCW pump 23</u>
A.	EDG 23	EDG 22
B.	EDG 21	EDG 23
C.	EDG 22	EDG 23
D.	EDG 21	EDG 22

Proposed Answer: D. EDG 21 EDG 22

Explanation:

Unit 2 Vital buses F, G, H are powered by EDG 23, 21, 22. Power supply to CCW pumps are F (pump 21), G (pump 22) and H (pump 23). Therefore, CCW pumps 22 and 23 are being powered by EDG 21 and 22 respectively

- A. Incorrect. Plausible, not a direct correlation of power supplies to loads. For instance CS pump 21 is powered from Bus G.
- B. Incorrect. Plausible, this is a power supply alignment for loads such as SI pumps.
- C. Incorrect. Plausible, this is a power supply alignment for loads such as AFW pumps.
- D. Correct. EDG 21 supplies bus G (pump 22) and EDG 22 supplies bus H (pump 23).

Technical References: OIM J-1-1

References to be provided to applicants during exam: None

Learning Objective: State the power supplies to CCW system components (8129)

Question Source: Bank # DCPD A-1102 X
(note changes; attach parent) Modified Bank #

Question History: New
Last Two NRC Exams

Question Cognitive Level: Memory/Fundamental X
Comprehensive/Analysis

10CFR Part 55 Content: 55.41.8

Difficulty: 2.0

Examination Outline Cross-Reference**010 K6.01 Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: Pressure detection systems**

Level	RO
Tier #	2
Group #	1
K/A #	010 K6.01
Rating	2.7

Question 09

Unit 1 is at 100% power.

Pressurizer pressure channel, PT-474, fails LOW.

If Pressurizer pressure rises to the open setpoint, ONLY Pressurizer PORV(s) _____ will open.

- A. PCV-474
- B. PCV-455C
- C. PCV-455C and PCV-456
- D. PCV-456

Proposed Answer: D. PCV-456

Explanation:

- A. Incorrect. PCV-474 is prevented from opening due to the low pressure interlock from PT-474. Plausible if its not known that PT-474 is the low pressure interlock channel for the control portion of the PORV circuitry which controls the PORV opening from the "second highest pressure" control circuitry. Also, logical to assume the interlock channel would be a different channel (as is the situation for PCV-455C)
- B. Incorrect. PCV-455C is prevented due to the failure of PT-474. Plausible because there is only one PORV that will open, but its PCV-456.
- C. Incorrect. Both PCV-455C and PCV-474 will not open. PCV-474 is independently controlled by its associated channel, PT-474. PCV-455C is prevented from opening due to low pressure interlock met due to the failed PT-474 less than 2185 psig. Plausible that PT-455C, with PCV-456 are unaffected by PT-474.
- D. Correct. 2 PORVs are prevented from opening but its PCV-455C and PCV-474. PCV-456 will still operate.

Technical References: OIM A-4-7**References to be provided to applicants during exam:** None**Learning Objective:** Discuss abnormal conditions associated with the Pzr, Pzr Pressure and Level Control System (36926)**Question Source:**

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Question History:

Last Two NRC Exams

No

Question Cognitive Level:

Memory/Fundamental

Comprehensive/Analysis

X

10CFR Part 55 Content:

55.41.7

Difficulty: 3.7

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	1
K/A #	012 K5.02
Rating	3.3

012 K5.02 -Knowledge of the operational implications of the following concepts as the apply to the RPS: Power Density

Question 10

Which of the following reactor trips is designed to prevent operation of the reactor with a power density of greater than 21.1 kW/foot?

- A. Power Range Rate (Positive) and Power Range High Flux (High)
- B. Overpower Delta T (OPΔT) and Over Temperature Delta T (OTΔT)
- C. Overpower Delta T (OPΔT) and Power Range High Flux (High)
- D. Power Range Rate (Positive) and Over Temperature Delta T (OTΔT)

Proposed Answer: C. Overpower Delta T (OPΔT) and Power Range High Flux (High)

Explanation:

- A. Incorrect. Power Range High Flux is correct, however, PR rate (positive) is for ejected rod (flux peaking)
- B. Incorrect. OPΔT is correct, OT is for DNB.
- C. Correct. OPΔT and Power Range High Flux are for excessive kw/foot (power density).
- D. Incorrect. Power Range High Flux is correct, however, OTΔT is DNB protection.

Technical References: OIM B-6-4-a

References to be provided to applicants during exam: None

Learning Objective: Analyze automatic features and interlocks associated with the Reactor Protection System. (37048)

Question Source:	Bank #36 DCPN NRC 07/2011 (L091)	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.2	

Difficulty: 3.0

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	1
K/A #	012 A1.01
Rating	2.9

012 A1.01 - Ability to predict and/or monitor Changes in parameters (to prevent exceeding design limits) associated with operating the RPS controls including: Trip setpoint adjustment

Question 11

Which of the following will cause the calculated setpoint for both Overpower Delta T (OPΔT) and Over Temperature Delta T (OTΔT) to LOWER?

1. RCS average temperature rising
 2. Major deviation in AFD
 3. RCS pressure lowering
- A. 1 ONLY
- B. 1 and 2
- C. 3 ONLY
- D. 2 and 3

Proposed Answer: A. 1 ONLY

Explanation:

OPΔT Set point adjusted lower by rising Tav_g only. Lowering pressure or changes in AFD will not affect the setpoint.

OTΔT Set point adjusted lower by rising Tav_g, lowering RCS pressure and a major deviation of AFD.

- A. Correct. OP ΔT is affected by Tav_g only. The AFD input is set to zero. OT ΔT would lower for lower RCS pressure but pressure does not affect OPΔT.
- B. Incorrect. This would be correct for OTΔT.
- C. Incorrect. Plausible because pressure is used in the OTΔT setpoint. But for DNB - lowering pressure would cause the setpoint to lower.
- D. Incorrect. Both have an input into OTΔT not OPΔT.

Technical References: LB-6A

References to be provided to applicants during exam: None

Learning Objective: Describe Reactor Protection System components. (7373)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.2	
Difficulty: 3.0		

Examination Outline Cross-Reference

013 G2.4.9 – ESFAS: Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Level RO
Tier # 2
Group # 1
K/A # 013 G2.4.9
Rating 3.8

Question 12

GIVEN:

- Unit 1 is performing a heatup in accordance with OP L-1, Plant Heatup From Hot Shutdown to Hot Standby
- Electrical power is aligned to startup
- RCS temperature is 525°F
- RCS pressure is 1900 psig
- PK08 indicates as shown below: (ONLY red outlined annunciators are lit)



A steam break, outside containment, upstream of the MSIV occurs on the 11 Steam Generator.

- 1) SI will _____ actuate.
 - 2) Once SI is automatically or manually actuated, the running AFW pumps will _____.
- A.
- 1) automatically
 - 2) remain running without interruption
- B.
- 1) NOT automatically
 - 2) remain running without interruption
- C.
- 1) automatically
 - 2) stop and restart when sequenced on to their respective Emergency Diesel Generator
- D.
- 1) NOT automatically
 - 2) stop and restart when sequenced on to their respective Emergency Diesel Generator

Proposed Answer: B. 1) NOT automatically
2) remain running without interruption

Explanation:

Tests what the operator would see in the Control Room (operational validity) when RCS pressure is less than 1915 psig and an accident occurs (steam break). Additionally, the response of the AFW pumps when the SI occurs (do they or don't they stop and sequence on when SI occurs).

- A. Incorrect. SI on low RCS pressure is blocked below P-11 (1915 psig) (PK08-06 LIT). Plausible if P-11 is thought to only affect RCS pressure SI. Also, SI would automatically actuate if the break was inside containment and pressure rises to greater than 3 psig.
- B. Correct. SI will not actuate automatically, however, the AFW are not stripped and then restarted. They would be if there was also a transfer to diesel, but if startup is available, no load stripping occurs.
- C. Incorrect. This would be the response at power and the AFW pumps were not running and startup was not available.
- D. Incorrect, P-11 blocks Low Pressurizer pressure AND low steamline pressure SI (PK08-16 and 17 LIT). The AFW are not stripped. Plausible because for most SI actuations which cause a reactor trip and bus transfer from 500 kV to either startup or diesel, the AFW pumps would be sequenced. In this case, already on startup, no bus stripping occurs and the pumps will remain running.

Technical References: OIM B-6-2, B-6-5 and J-6-1

References to be provided to applicants during exam: None

Learning Objective: 41313 - Analyze automatic features and interlocks associated with the Eagle-21/SSPS

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #11 DCPN NRC L091C 03/12

X

New

Past NRC Exam

Yes

Question History:

Last Two NRC Exams

No

Question Cognitive Level:

Memory/Fundamental

Comprehensive/Analysis

X

10CFR Part 55 Content:

55.41.7

Difficulty: 3.3

Examination Outline Cross-Reference

013 A2.05 Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations; Loss of dc control power

Level	RO
Tier #	2
Group #	1
K/A #	013 A2.05
Rating	3.7

Question 13

A loss of DC Bus 22 causes the reactor to trip from 100% power.

In accordance with OP AP-23, Loss of Vital DC Bus;

- 1) 21 Turbine Driven AFW pump Steam Isolation valve, FCV-95, will have _____.
 - 2) 21 Turbine Driven AFW pump motor operated valves, LCV's 106, 107, 108 and 109 _____.
- A. 1) failed open
2) can be operated from the Control Room
- B. 1) failed open
2) will have to be operated locally
- C. 1) to be opened locally
2) can be operated from the Control Room
- D. 1) to be opened locally
2) will have to be operated locally

Proposed Answer: D. 1) to be opened locally
2) will have to be operated locally.

Explanation:

Per OP AP-23,

If the Turbine Driven AFW pump 2 1 is needed:

FCV 95, Steam Isolation valve, will have to be manually opened.

Turbine Driven AFW pump motor operated valves LCV 106, 107, 108, and 109 will be deenergized if 480V bus G is lost and personnel will have to be dispatched locally to control AFW flow. .

- A. Incorrect. DC bus 22 powers FCV-95 and it will not open. Additionally, bus G will be deenergized and the LCVs, which are powered from 480 VAC bus G will not have power. Plausible because valves have different fail position, and it could be thought FCV-95 fails open (to ensure start of the AFW pump) or because the LCVs are MOVs the loss of DC is not obvious that it prevents valve operation..
- B. Incorrect. FCV-95 fails "as is" on a loss of DC. Plausible because many DC components fail open/closed on a loss of power.
- C. Incorrect. FCV-95 fails "as is" on a loss of DC. Plausible because many DC components fail open/closed on a loss of power.
- D. Correct. FCV-95 has lost DC power and fails as-is, it must be opened locally. The LCVs are also not powered (are in the full open position) due to the loss of the 480 VAC bus and must be operated locally.

Technical References: OP AP-23 appendix B

References to be provided to applicants during exam: None

Learning Objective: Explain the consequences of loss of DC vital bus. (7116)

Question Source:	Bank #DCPP Bank S-27165	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41)7	
Difficulty: 2.0		

L162 Answer Key

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	1
K/A #	022 A3.01
Rating	4.1

022 A3.01 Ability to monitor automatic operation of the CCS, including: Initiation of safeguards mode of operation

Question 14

Safety Injection occurs on Unit 1. 4 kV Bus G is de-energized when the reactor trips.

The operator is performing Appendix E, ESF Auto Actions, Secondary and Auxiliaries Status, of E-0, Reactor Trip or Safety Injection, and notes that the Monitor Light Box C white lights for CFCUs 12 and 15 are lit.

The white light is ___ 1) ___ for CFCU 12 and ___ 2) ___ for CFCU 15.

- A. 1) expected 2) expected
- B. 1) unexpected 2) expected
- C. 1) expected 2) unexpected
- D. 1) unexpected 2) unexpected

Proposed Answer: B. 1) unexpected 2) expected

Explanation:

Safety Injection starts all CFCUs in LOW speed. When the CFCUs start, the white status light goes out to indicate it is running in the "safeguards" mode. If light is lit, the CFCU is not running or is still in high speed.

- A. Incorrect. CFCU 12 has power and should have shifted to LOW (light should be out). Plausible if the power supplies are not known, but it is known that there are 2 CFCUs on bus G.
- B. Correct. CFCU 12 should be running and is not as indicated by the white light (unexpected). CFCU 15 is powered from bus G and is de-energized (expected light lit).
- C. Incorrect. CFCU 15 does not have power and will not be running (expected). CFCU 12 should be running in LOW. (unexpected). This is plausible if power supplies are mixed.
- D. Incorrect. CFCU 15 does not have power (expected) but CFCU 12 does and the light should be out. Plausible if the power supplies are not known but it is known that there are 2 CFCUs on Bus G and is a lineup of CFCUs such as 13 and 14..

Technical References: E-0 appendix A and E

References to be provided to applicants during exam: None

Learning Objective: Discuss abnormal conditions associated with the CFCUs. (40812)

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Question History:

Last Two NRC Exams

No

Question Cognitive Level:

Memory/Fundamental

Comprehensive/

X

10CFR Part 55 Content:

Analysis 55.41.7

Difficulty: 2.0

Examination Outline Cross-Reference

026 A4.01 Ability to manually operate and/or monitor in the control room: CSS controls

Level	RO
Tier #	2
Group #	1
K/A #	026 A4.01
Rating	4.5

Question 15

GIVEN:

- A large break LOCA has occurred
- Containment Spray has actuated

The Containment Spray pumps will _____ 1) _____ when RWST level reaches _____ 2) _____

- A. 1) automatically trip 2) 4%
- B. 1) automatically trip 2) 33%
- C. 1) be stopped by the operator 2) 4%
- D. 1) be stopped by the operator 2) 33%

Proposed Answer: C. 1) be stopped by the operator 2) 4%

Explanation:

- A. Incorrect. The RHR pumps automatically trip (at 33%) RWST level. The CS pumps are left to run until 4% level, however, they must be stopped by the operator. There is not an auto trip on level.
- B. Incorrect. The RHR pumps automatically trip (at 33%) RWST level. The CS pumps are left to run until 4% level, however, they must be stopped by the operator. There is not an auto trip on level..
- C. Correct. At 4%, the operator is instructed to stop the CS pumps.
- D. Incorrect. The operator stops the pumps, but only after RWST reaches 4%. 33% is the level of switching to cold leg recirc (and RHR pumps trip). Plausible to think the CS pumps are stopped until the RHR pumps are restarted and realigned to provide spray.

Technical References: LB-6A AR PK03-05, E-1.3, LI-2

References to be provided to applicants during exam: None

Learning Objective: 37578 - Describe controls, indications, and alarms associated with the CSS

Question Source:	Bank	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Difficulty: 3.7

Examination Outline Cross-Reference

039 K3.05 Knowledge of the effect that a loss or malfunction of the MRSS will have on the following: RCS

Level	RO
Tier #	2
Group #	1
K/A #	039 K3.05
Rating	4.1

Question 16

The plant is operating at 35% power. Control Rods are in MANUAL.

Main Steam Isolation Valve (MSIV), FCV-41, closes.

Assuming the plant does NOT trip:

- 1) ΔT in RCS loop 11 will _____.
 - 2) ΔT in RCS loops 12, 13 and 14 will _____.
- A. 1) rise
2) lower
 - B. 1) lower
2) rise
 - C. 1) rise
2) remain the same
 - D. 1) lower
2) remain the same

Proposed Answer: B. 1) lower
2) rise

Explanation:

- A. Incorrect. Loop 11 steam flow will go toward 0. As a result, there will be very little heat removal from the loop and loop ΔT will lower. The increased steam demand from the other 3 loops will cause Tcold in the other 3 loops to lower and ΔT will rise. This answer is the opposite of what occurs.
- B. Correct. Loop 11 steam flow will go toward 0. As a result, there will be very little heat removal from the loop and loop ΔT will lower. The increased steam demand from the other 3 loops will cause Tcold in the other 3 loops to lower and ΔT will rise.
- C. Incorrect. RCS loop 11 ΔT will lower as heat removal from that generator lowers. Reactor power, will not change appreciably, however, loop ΔT for the remaining loops will rised .
- D. Incorrect. It is correct that loop ΔT will lower, however, while reactor power will remain approximately the same, the other loops ΔT will rise as the same steam demand is still being delivered to the turbine.

Technical References: TH18T pages 30 - 32

References to be provided to applicants during exam: None

Learning Objective: DESCRIBE the primary and secondary plant responses to specific transients. (10579)

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #DCPP bank F-22216

New

X

Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
10CFR Part 55 Content:	Comprehensive/Analysis	X
Difficulty: 2.3	55.41.5	

L162 Answer Key

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	1
K/A #	059 K4.02
Rating	3.3

059 K4.02 - Knowledge of MFW design feature(s) and/or interlock(s) which provide for the following: Automatic turbine/reactor trip runback.

Question 17

Unit 2 is at 100% power.

Main Feedwater pump 21 trips.

Turbine load will begin to lower at a rate of _____ 1) _____ and continue at this rate until turbine load reaches _____ 2) _____.

- A. 1) 225 MW/min 2) 550 MW
- B. 1) 225 MW/min 2) 650 MW
- C. 1) 532 MW/min 2) 550 MW
- D. 1) 532 MW/min 2) 650 MW

Proposed Answer: B. 1) 225 MW/min 2) 650 MW

Explanation:

- A. Incorrect. Ramp at 225 MW/min is to 650 MW. Plausible – this is the target for the trip of a circ water pump, additionally, the MFW pump ramp continues from 650 to 550 MW at a rate of 25 MW/min.
- B. Correct. A program ramp occurs at a rate of 225 MW/min down to 650 MW (and then lower to 550 MW at 25 MW/min).
- C. Incorrect. 532 MW/min is the rate for the trip of a stator water cooling pump and changes to 25 MW/min at 650 until 550 MW is reached.
- D. Incorrect. 532 MW/min is the rate for a stator cooling water low pressure runback. 650 MW is the setpoint for the initial decrease, 530 MWe is the end of the ramp, (650 to 550 MW at a rate of 25 MW/min).

Technical References: LC-3C

References to be provided to applicants during exam: None

Learning Objective: Analyze automatic features and interlocks associated with the DEHC System. (5561)

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Question History:

Last Two NRC Exams

No

Question Cognitive Level:

Memory/Fundamental

X

Comprehensive/Analysis

10CFR Part 55 Content:

55.41.7

Difficulty: 2.0

Examination Outline Cross-Reference

061A1.04 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the AFW controls including: AFW source tank level.

Level	RO
Tier #	2
Group #	1
K/A #	061 A1.04
Rating	3.9

Question 18

The crew is performing EOP E-0, Reactor Trip or Safety Injection.

CST level is 39% and lowering at a rate of 10% an hour.

In _____ hour(s) the CST will reach the minimum allowable level by the EOP and the crew will be required to align the AFW pump suction to an alternate source.

- A. less than 1
- B. approximately 2.5
- C. approximately 3
- D. approximately 3.5

Proposed Answer: C. approximately 3

Explanation:

All levels correspond to EOP setpoints.

- A. Incorrect. In less than one hour level will be less than 33% which if the swapover setpoint for the RWST to the containment sump.
- B. Incorrect. 15% is the level used in the EOPs for the steam generators will be reached in approximately 2.5 hours). CST swap is performed when level is less than 10% (approximately 3 hours).
- C. Correct. The level to transfer is 10%, in 3 hours, level will be less than 10% and transfer to an alternate source is required.
- D. Incorrect. In 3.5 hours, level will be almost 4% which is the RWST level the containment spray pumps are manually tripped.

Technical References: E-0, E-1, E-1.3

References to be provided to applicants during exam: None

Learning Objective:

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Question History:

Last Two NRC Exams

No

Question Cognitive Level:

Memory/Fundamental

Comprehensive/Analysis

X

10CFR Part 55 Content:

55.41.4

Difficulty: 2.3

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	1
K/A #	061 K1.01
Rating	4.1

061 K1.01 Knowledge of the physical connections and/or cause-effect relationships between the AFW and the following systems: S/G system

Question 19

A steamline break causes the plant to trip from 100% power.

3 minutes later steam generator pressures are:

- 11 – 900 psig, stable
- 12 – 500 psig, lowering
- 13 – 500 psig, lowering
- 14 – 870 psig, stable

Which of the following is a possible location for the steam break that caused the plant to trip?

- A. Upstream of Steam Generator 12 MSIV, FCV-42
- B. Downstream of Steam Generator 13 MSIV, FCV-43
- C. Upstream of steam inlet valve to the TDAFW pump, FCV-95
- D. Downstream of steam inlet valve to the TDAFW pump, FCV-95

Proposed Answer: C. Upstream of steam inlet valve to the TDAFW pump, FCV-95

Explanation:

- A. Incorrect. Both 12 and 13 Steam Generator pressures are low. Check valves between the generators (off the AFW pump line) prevent both from depressurizing.
- B. Incorrect, MSIVs are shut which would have isolated the break if this was the location.
- C. Correct. Both steam generators feed the AFW pump. A break on the line upstream of the FCV would cause both to depressurize and cause the reactor trip.
- D. Incorrect. The plant tripped due to the steam break. FCV-95 is normally closed. A break here would not have affected plant operation at power

Technical References: OVID 106704 sheet 4

References to be provided to applicants during exam: None

Learning Objective: 7240 - Identify the location of Main Steam system valves and piping

Question Source:	Bank #45 DCPD NRC Exam L091 07/2011	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	
Difficulty: 2.3		

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	1
K/A #	062 A2.03
Rating	2.9

062 A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Consequences of improper sequencing when transferring to or from an inverter.

Question 20

Unit 1 is at 100% power. A Nuclear Operator has been dispatched to return IY-12 to its normal supply and then place IY-13 on its backup supply.

Multiple seemingly unrelated alarms including the following indications are noted in the control room;

- Channel II bistable status lights - ON
- Channel III bistable status and power supply lights – OFF

- 1) Annunciator window _____ will be lit.
- 2) The crew should enter OP AP-4, Loss of Vital or Non-Vital Instrument AC, and perform _____ to take steps to address the problem.
 - A. 1) PK19-18, VITAL UPS TROUBLE
2) Section C, "Loss of PY-12/PY-22"
 - B. 1) PK19-18, VITAL UPS TROUBLE
2) Section D, "Loss of PY-13/PY-23"
 - C. 1) PK19-19, VITAL UPS FAILURE
2) Section C, "Loss of PY-12/PY-22"
 - D. 1) PK19-19, VITAL UPS FAILURE
2) Section D, "Loss of PY-13/PY-23"

Proposed Answer: C. 1) PK19-19, VITAL UPS FAILURE 2) Section C, "Loss of PY-12/PY-22"

Explanation:

Instrument Panel Lost	Bistable Status
PY 11 (21)	
Channel I bistable status lights	ON
IV power supply (if due to the loss of PY-11A)	
PY 12 (22)	
Channel II bistable status lights	ON
Channel III bistable status and power supply lights	OFF
PY 13 (23)	
Channel III bistable status lights	ON
PY 13A (23A)	

Channel IV bistable status and power supply lights OFF

PY 14 (24)

Channel IV bistable status lights ON

Channel I bistable status and power supply lights OFF

- A. Incorrect. The Failure PK will be lit – however, PY-12 is the affected instrument bus
- B. Incorrect. The failure PK will be lit and the affected PY is PY-12, not 13.
- C. Correct. Indications are that a PY has lost power. PY-12 bistables trip and the power supply light for channel III go out if PY-12 is de-energized.
- D. Incorrect. The Failure PK will be lit however, the PY affected is PY-12 not PY-13.

Technical References: OP AP-4. AR PK-19-18, AR PK19-19

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:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Question History:

Last Two NRC Exams

No

Question Cognitive Level:

Memory/Fundamental

Comprehensive/Analysis

X

10CFR Part 55 Content:

55.41.7

Difficulty: 2.7

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	1
K/A #	062 K4.03
Rating	2.8

062 K4.03 Knowledge of ac distribution system design feature(s) and/or interlock(s) which provide for the following: Interlocks between automatic bus transfer and breakers

Question 21

Unit 1 is in MODE 3 following a reactor trip. Startup power is supplying all AC buses.

A loss of Startup power occurs. The crew is now taking action to backfeed from the 500 kV system in accordance with OP J-2:V, Backfeeding the Unit From the 500kV System.

Which of the following actions must be taken by the operator before the Main Generator Output circuit breakers, CB-532 and CB-632, can be closed?

- A. Reset the 4 kV and 12 kV auto transfers.
- B. Close the Motor Operated Disconnect from CC3.
- C. Reset the Unit Lockout Relays, 86G1 and 86G11 on VB4.
- D. Switch all device CUTOOUT switches on VB4 with Blue lamacoids from CUTOOUT to CUTIN.

Proposed Answer: C. Reset the Unit Lockout Relays, 86G1 and 86G11 on VB4

Explanation:

- A. Incorrect. the auto transfer relays do not input into the operation of the 500 kV breaker. Plausible because the goal of backfeeding is to supply those buses from 500 kV, it could be that the auto transfers would be a block in closing in that supply to the buses.
- B. Incorrect. The MOD must be OPEN, not closed.
- C. Correct. Tripping the 86G1 and G11 relays trips 532 and 632. they must be reset before the breakers can be closed.
- D. Incorrect. The CUTOOUT switches are taken to CUTOOUT. The normal (pretrip) position of these switches is CUTIN..

Technical References: OP J-2:V, LJ-4A

References to be provided to applicants during exam: None

Learning Objective: 5280 - Analyze automatic features and interlocks associated with the Main Generator.

Question Source:	Bank #20 DCPD NRC Exam L091C 03/2012	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Difficulty: 2.7

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	1
K/A #	063 A2.01
Rating	2.5

**063 A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Grounds**

Question 22

Unit 1 is at 100% power.

A ground causes the DC breaker supplying control power to the 11 RCP, 13 RCP and 12 CWP to open.

All indicating lights for 11 RCP, 13 RCP and 12 CWP are out.

In accordance with AR PK05-01, 11 RCP, or AR PK05-03, 13 RCP, what action(s) should be taken by the crew in the Control Room?

- 1) Dispatch an operator to check a possible ground and the status of the DC control power breaker to the affected components on 12 kV bus D
 - 2) Dispatch an operator to check a possible ground and the status of the DC control power breaker to the affected components on 12 kV bus E
 - 3) From the Control Room trip the reactor, 11 RCP, 13 RCP and 12 CWP
- A. 1 ONLY
- B. 2 ONLY
- C. 1 and 3
- D. 2 and 3

Proposed Answer: B. 2 ONLY

Explanation:

NOTE: the opening of the DC control power breaker causes a loss of indication for RCPs 11, 13 and CWP 12. It also causes a loss of indication for the redundant breakers for the other two RCPs which does not affect the Control Room indication. The RCP alarms for the other two RCPs is received, however, there is no loss of indication in the Control Room. They are left out of the information given as it is not pertinent to the action to be taken.

- A. Incorrect. A condition, such as a ground has caused a loss of control power, which causes the lights to go out, however, the loads are on Bus E not bus D.
- B. Correct. A condition, such as a ground has caused a loss of control power, which causes the lights to go out. The three loads listed are on 12 kV bus E. Per PK05-01 or 05-03, the action is to dispatch an operator to check the DC control power breaker, 72-1233, "12KV SWGR BUS E & SVD6R, SVD7R (the lights for the RCP redundant breakers)
- C. Incorrect. Loads are on Bus D. There is no reason to trip the loads and due to the loss of control power and they cannot be tripped from the control room. Plausible because there is no indication and it could be thought the components need to be tripped. If the RCPs were required to be tripped, the reactor is always tripped first.
- D. Incorrect. There is no reason to trip the loads and due to the loss of control power and they cannot be tripped from the control room. Plausible because there is no indication and it

could be thought the components need to be tripped. If the RCPs were required to be tripped, the reactor is always tripped first.

Technical References: AR PK05-01, 05-03, drawings 445075, 477848

References to be provided to applicants during exam: None

Learning Objective: 37793 -Describe controls, indications, and alarms associated with the DC Power System

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content: Difficulty: 2.7	55.41.10	

L162 Answer Key

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	1
K/A #	064 K4.01
Rating	3.8

064 K4.01 Knowledge of ED/G system design feature(s) and/or interlock(s) which provide for the following: Trips while loading the ED/G (frequency, voltage, speed)

Question 23

A loss of all offsite power has occurred.

30 seconds after EDG 12 automatically starts and begins loading, it trips.

What condition(s), individually, could have caused EDG 12 to trip?

- 1) Differential
 - 2) Overspeed
 - 3) High Jacket Water temperature
- A. 2 ONLY
- B. 3 ONLY
- C. 1 and 2
- D. 1 and 3

Proposed Answer: C. 1 and 2

Explanation:

The shutdown relay (SDR) is energized by any of the following:	And isolates the Diesel by:
<ul style="list-style-type: none">• Engine mechanical overspeed trip (1050 rpm)• Low lube oil pressure (< 40 psig)<ul style="list-style-type: none">• 20-seconds time delay on start• Emergency trip PB is depressed• Diesel generator differential relay actuates• High jacket water temperature (205°F)*• Diesel overcrank (> 10 seconds)*	<ul style="list-style-type: none">• Tripping the diesel engine• Deenergizing generator excitation• Tripping the diesel output breaker

* These SDR trips are bypassed unless the diesel is in local control.

Note: no trips of EDGs specifically associated with frequency, (KA).

- A. Incorrect. Overspeed will cause the diesel to trip but Differential will, as well. Plausible as there are trips that are normally cut out and it could thought one of these is normally cutout.
- B. Incorrect. High Jacket Water temperature is only active when control is in LOCAL. If the EDG was started in the Control Room, Local/Remote Control Selector switch is in Remote..
- C. Correct. Overspeed could cause the trip, as will differential.
- D. Inconrret. Differential will but the high jacket water temperature is not a possible trip at this time (only active when in LOCAL).

Technical References: LJ-6B

References to be provided to applicants during exam: None

Learning Objective: Analyze automatic features and interlocks associated with the Diesel Generator System. (37725)

Question Source:
(note changes; attach parent)

Bank #
Modified Bank #
New

X

Question History:

Last Two NRC Exams

No

Question Cognitive Level:

Memory/Fundamental
Comprehensive/Analysis

X

10CFR Part 55 Content:
Difficulty: 2.3

55.41.7

L162 Answer Key

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	1
K/A #	064 K6.07
Rating	2.7

064 K6.07 Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Air receivers

Question 24

GIVEN:

- Unit 1 is at 100% power
- A relief valve fails open on Starting Air Receiver “A” for Emergency Diesel Generator 11 and the air receiver begins to rapidly depressurize
- The leakage exceeds the capacity of the starting air compressor

A loss of offsite power occurs.

Which of the following describes the response of Emergency Diesel Generator 11?

- A. The diesel will not start due to a loss of control air to fuel oil day tank LCVs.
- B. The diesel will not start because the fuel rack booster pump will not be supplied with sufficient air for operation.
- C. The diesel will start in the normal time via the two starting air solenoids associated with the Starting Air Receiver “B”.
- D. The diesel will start in the normal time via all four starting air solenoids cross-tied to Starting Air Receiver “B”.

Proposed Answer: C. The diesel will start in the normal time via the two starting air solenoids associated with the Starting Air Receiver “B”

Explanation:

- A. Incorrect. The LCVs are provided with air from the air receivers to operate. However, the trains are not cross-tied. Train A and B each supply air to a day tank LCV. One train will/could be compromised but the system would still operate from the other train. Plausible because systems, such as CCW are cross tied to ensure cooling to both A and B headers.
- B. Incorrect. Plausible because air is the force to move the booster pump to the max fuel position on start up. However, they are not cross tied and the other train would operate.
- C. Correct. The systems are not cross tied and only one train (each 100% capacity) is required to start..
- D. Incorrect. If the systems were cross-tied this could be true, however, they are not (there is a manual cross-tie that is maintained closed).

Technical References: LJ-6B

References to be provided to applicants during exam: None

Learning Objective: 6431 - State the purpose of D/G subsystems and components

Question Source:	Bank #19 DCPN NRC (L111) Exam 11/2012	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No

Question Cognitive Level:

Memory/Fundamental
Comprehensive/Analysis

X

10CFR Part 55 Content:

55.41.8

Difficulty: 2.0

L162 Answer Key

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	1
K/A #	073 K1.01
Rating	3.6

073 K1.01 - Knowledge of the physical connections and/or cause effect relationships between the PRM system and the following systems: Those systems served by PRMs

Question 25

Unit 2 is at 100% power.

A small steam generator tube leak is causing steam line radiation monitor RM-73 to read 1000 cpm.

If power is reduced to 50%, indication on RM-73 will:

- A. lower due to a decrease in N-16 production.
- B. lower due to a decrease in I-131 production.
- C. rise because of the rise in Xenon production.
- D. rise because there is less steam flow but the same amount of radiation.

Proposed Answer: A. lower due to a decrease in N-16 production.

Explanation:

- A. Correct. The steam line radiation monitors detect N-16 from the tube leakage, as power is lowered, N-16 production lowers and the reading on RM-73 lowers. Once the unit is shutdown, N-16 production ceases and the indication will decrease
- B. Incorrect. Iodine production will decrease but the detector senses N-16 not iodine (or xenon)
- C. Incorrect. Xenon production initially will rise as power lowers, however, the rad monitors measure N-16 and its production is lowering as power lowers.
- D. Incorrect. A lower flowrate of steam could affect air ejector reading but the steamline rad monitors are not in the flow stream but outside the piping and measure N-16

Technical References: LG4A – Radiation Monitoring, SOE-93-001

References to be provided to applicants during exam: None

Learning Objective: 8485 Explain the conditions that effect Radiation Monitoring system radiation monitor indications

Question Source: (note changes; attach parent)	Bank #51 DCPN NRC Exam 02/2005	X
	Modified Bank #	
	New	
	Previous NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.11	
Difficulty: 2.7		

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	1
K/A #	076 K2.01
Rating	2.7

**076 K2.01 Knowledge of bus power supplies to the following:
Service water**

Question 26

A loss of offsite power occurs on Unit 2.

What Emergency Diesel Generators (EDG) will supply power to the Unit 2 Auxiliary Saltwater Pumps, 2-1 and 2-2?

<u>ASW Pump 2-1</u>	<u>ASW Pump 2-2</u>
A. EDG 2-1	EDG 2-2
B. EDG 2-2	EDG 2-3
C. EDG 2-3	EDG 2-2
D. EDG 2-3	EDG 2-1

Proposed Answer: D. EDG 2-3 EDG 2-1

Explanation:

- A. Incorrect. Pumps 1 and 2 are powered from Bus F and G respectively. The EDGs for those buses are 2-3 and 2-1 (buses G and H). Its plausible that EDG 1 and 2 would supply pumps 1 and 2.
- B. Incorrect. Pumps 1 and 2 are powered from Bus F and G respectively. Plausible because there are pumps on Bus G and Bus H (containment spray pumps) and ASW cross tie valves, FCV-495 and 496 are powered by Bus H.
- C. Incorrect. Plausible because there are pumps on Bus F and Bus H (AFW pumps) and cross tie valves, FCV-495 and 496 are powered by Bus H.
- D. Correct. Pumps 1 and 2 are powered from Bus F and G respectively. EDGs 2-3 and 2-1 power those buses.

Technical References: OIM J-1-1

References to be provided to applicants during exam: None

Learning Objective: State the power supplies to ASW System components. (5339)

Question Source:	Bank	
(note changes; attach parent)	Modified Bank #26 DCPN NRC Exam, (L051)	X
	04/07	
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.4	

Difficulty: 2.0

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	1
K/A #	078 A3.01
Rating	3.1

078 A3.01 Ability to monitor automatic operation of the IAS, including: Air pressure

Question 27

All instrument air compressors are available and are aligned in a normal system alignment.

Instrument air header pressure lowers from 106 to 99 psig.

What will be the status of the instrument air compressors?

- A. All compressors are running and loaded.
- B. Only Rotary compressors 0-5, 0-6 and 0-7 are running and loaded.
- C. Only Reciprocating compressors 0-1 through 0-4 are running and loaded.
- D. All compressors running but only Rotary compressors 0-5 and 0-6 are loaded.

Proposed Answer: B. Only Rotary compressors 0-5, 0-6 and 0-7 are running and loaded.

Explanation:

- A. Incorrect. For a normal system configuration, the reciprocating air compressors will start when pressure is 93 psig and the rotary air compressors start at 103 psig.
- B. Correct. The rotary air compressors (0-5, 6 and 7) start at 103 psig and the reciprocating compressors will start at 93 psig.
- C. Incorrect. The rotary air compressors start first, followed by the reciprocating.
- D. Incorrect. At this pressure, all compressors will be loaded. If pressure was higher, its possible only a compressor could be running unloaded.

Technical References: OIM K-1-2

References to be provided to applicants during exam: None

Learning Objective: 37565 Analyze automatic features and interlocks associated with the Compressed Air System.

Question Source:	Bank #27 DCPD NRC Exam 03/2012	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	
Difficulty: 2.7		

Examination Outline Cross-Reference

103 A4.04 Ability to manually operate and/or monitor in the control room: Phase A and phase B resets.

Level	RO
Tier #	2
Group #	1
K/A #	103A4.04
Rating	3.5

Question 28

GIVEN:

- Phase A Red lights on Monitor Light Box B - LIT
- Phase B Red lights on Monitor Light Box D - LIT
- PK02-01, CONTMT ISOLATION PHASE A/B - LIT
- PK02-02, SAFETY INJECTION INITIATE - LIT
- PK08-21, SAFETY INJECTION ACTUATION - LIT
- Containment pressure is 10 psig

The operator presses the RESET pushbuttons for Phase A and Phase B.

After the Phase A Reset pushbuttons are pressed, the Phase A Red lights will be:

After the Phase B Reset pushbuttons are pressed, the Phase B Red lights will be:

	<u>Phase A</u>	<u>Phase B</u>
A.	Lit	Lit
B.	Not Lit	Lit
C.	Lit	Not Lit
D.	Not Lit	Not Lit

Proposed Answer: D. Not Lit Not Lit

Explanation:

Indications in the initial conditions are that:

- SI has actuated and not reset (PK02-02 and 08-21).
 - Either phase A or B signal cause PK02-01 to light.
- A. Incorrect. The resets for both signals are "retentive" memory and will reset with the actuating signal present or SI is not reset or containment pressure is above the actuation setpoint (currently 10 psig, setpoint is 3 psig). Plausible as some signals, such as FWI cannot be reset if containment pressure is high or there are failures, such as P-4 that would result in only partial SI reset and the lights/alarms could still be in.
- B. Incorrect. Phase A will reset and the red light will go out. Phase B will also reset. It does not need SI reset or containment pressure to be less than the actuating setpoint.
- C. Incorrect. Phase A does not need SI reset (although SI generates Phase A) or containment pressure less than 3 psig. Phase B will reset and the light will go out, there is not a need to reset SI.
- D. Correct. Both will reset even with SI not reset.

Technical References: OIM B-6-5, B-6-6, B-6-7, B-6-8

References to be provided to applicants during exam: None

Learning Objective: 37048 Analyze automatic features and interlocks associated with the RPS

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	
Difficulty: 3.0		

L162 Answer Key

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	2
K/A #	002 K1.07
Rating	3.5

002 K1.07 Knowledge of the physical connections and/or cause-effect relationships between the RCS and the following systems: Reactor vessel level indication system

Question 29

A natural circulation cooldown is being performed on Unit 1 in accordance with EOP E-0.3, Natural Circulation Cooldown with Steam Void in the Reactor Vessel (with RVLIS).

Per the background document for E-0.3, why must RVLIS Full Range indication be maintained greater than 69.2% during the cooldown?

- A. To maintain conditions for an RCP start should power become available.
- B. To ensure the core remains covered in the event T-hot reaches saturated conditions.
- C. To prevent steam from entering the hot leg and disrupting the natural circulation flow.
- D. To ensure that when the void in the upper head is collapsed, pressurizer level will remain on scale.

Proposed Answer: C. To prevent steam from entering the hot leg and disrupting the natural circulation flow

Explanation:

Question tests the knowledge of the RCS hot leg level that correlates to a RVLIS indication and the "cause/effect" if RVLIS is less than that level.

- A. Incorrect. A continuous action in the natural circulation cooldown procedures is to start an RCP.
- B. Incorrect. This is a non-accident procedure, core cooling is not a concern.
- C. Correct. Purpose of the step, as stated in the background is "To allow the void to reach the hot legs without disrupting natural circulation".
- D. Incorrect. If RVLIS is less than 100%, Pressurizer level is controlled to accommodate void collapse if an RCP is started.

Technical References: E-0.3, E-0.3 background

References to be provided to applicants during exam: None

Learning Objective: 5855 - Explain the effect of voids on RCS operation

Question Source: (note changes; attach parent)	Bank #26 DCPN NRC Exam 07/2011 Modified Bank # New Past NRC Exam	X Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content: Difficulty: 3.3	55.41.10	

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	2
K/A #	014 K4.05
Rating	3.1

014 K4.05 -Knowledge of RPIS design feature(s) and/or interlock(s) which provide for the following: Rod hold interlocks

Question 30

GIVEN:

- Unit 1 has been at 100% power for 300 days
- Control Bank D rods are at 222 steps on DRPI

The crew is placing Charging pump 11 in service. Charging pump 11 was last run 60 days ago.

Which of the following is the expected plant response when the operator starts Charging pump 11?

- A. Dilution; rods will not respond no matter how much Tave rises.
- B. Boration; rods will not respond no matter how much Tave lowers.
- C. Dilution; rods will step if Tave rises by more than 1.5°F.
- D. Boration; rods will step if Tave lowers by more than 1.5°F.

Proposed Answer: B. Boration; rods will not respond no matter how much Tave lowers

Explanation:

C-11 - Rod stop interlock is 220 steps.

- A. Incorrect. Rods will not move due to C-11. The expected change is boration (RCS concentration will have gone down since the pump was last run).
- B. Correct. Reactivity changes may result from placing idle portions of the CVCS in service due to differences in boron concentration in the piping. Boron concentration in the piping will be higher than current RCS concentration, as boron concentration lowers for most of core life. This will cause RCS temperature to lower. If Tave changes by more than 1.5F, rods would step out, however, at full power, rods will be above C-11 and prevent any further rod motion in auto.
- C. Incorrect. Boration will occur. A dilution would cause rod motion if temperature changes by at least 1.5°F
- D. Incorrect. Boration will occur, however, rods will not move due to C-11. This would be the correct answer if rods were less than 220 steps.

Technical References: OP B-1A:V, OIM B-6-3

References to be provided to applicants during exam: None

Learning Objective:

Question Source: (note changes; attach parent)	Bank #2 DCPD NRC 8/2014 Modified Bank # New Past NRC Exam	X Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 X
10CFR Part 55 Content: Difficulty: 2.7	55.41.2	

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	2
K/A #	015 A1.05
Rating	3.1

015 A1.05 Ability to predict and/or monitor changes in parameters to prevent exceeding design limits) associated with operating the NIS controls including: Imbalance (axial shape)

Question 31

Unit 1 has been at 100% power for 200 days following a refueling.

The crew is preparing to ramp from 100% to 50% using OP L-4, Normal Operation at Power. AFD is on target at -2.0.

- 1) At 50%, AFD, if maintained on target should be _____ negative.
- 2) Per OP L-4, the rate of ramp reduction will be limited to 5 MW/min to:
 - A. 1) less 2) minimize the effects of xenon oscillations on axial offset
 - B. 1) more 2) minimize the effects of xenon oscillations on axial offset
 - C. 1) less 2) prevent exceeding fuel conditioning limits
 - D. 1) more 2) prevent exceeding fuel conditioning limits

Proposed Answer: A. 1) less 2) minimize the effects of xenon oscillations on axial offset

Explanation:

NOTE: fuel is preconditioned at a given power level, once the plant has operated at or above that power level for at least 72 cumulative hours out of the last 7 days (168 hours). At 200 days, the fuel is preconditioned.

- A. Correct. Per OP L-4, P&L 6.2.2.f, When all fuel-related limitations are removed, an overall administrative restriction will limit load ramp rates to 5 megawatts per minute. This limit is imposed to reduce the effects of xenon oscillations on axial offset. At target AFD should be -1.0 (less negative)
- B. Incorrect. Reason is correct. Effect of power reduction on AFD is reversed.
- C. Incorrect. AFD is correct. There are restrictions on fuel conditioning but after 200 days at full power, the fuel is conditioned and the restrictions will not apply.
- D. Incorrect. Target and reason are incorrect as stated above.

Technical References: LB-4, OP L-4

References to be provided to applicants during exam: None

Learning Objective: Apply fundamental topics associated with the Excore Nuclear Instrumentation System. (7610)

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Past NRC Exam

No

Last Two NRC Exams

No

Question History:**Question Cognitive Level:**

Memory/Fundamental

Comprehensive/Analysis

X

10CFR Part 55 Content:

55.41.1

Difficulty: 2.7

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	2
K/A #	033 A2.02
Rating	2.7

033 A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the Spent Fuel Pool Cooling System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of SFPCS

Question 32

Unit 1 is at full power.

A loss of Spent Fuel Pool cooling has occurred. Spent Fuel Pool temperature is 208°F and rising. The crew is performing OP AP-22, Spent Fuel Pool Abnormalities.

In accordance with OP AP-22, Appendix A, "Addition of Water to the Spent Fuel Pool", what is the preferred source of makeup?

- A. Refueling Water Storage Tank
- B. Ionics
- C. Liquid Holdup Tank
- D. Condensate Storage Tank

Proposed Answer: B. Ionics

Explanation:

- A. Incorrect. This is the preferred source if the addition is to makeup for leakage and boron must be added. For the current conditions, makeup is for evaporation.
- B. Correct. Makeup is for evaporation (step 17 RNO). Per appendix A, the preferred unborated source is Ionics.
- C. Incorrect. This is a method in appendix A to add borated makeup.
- D. Incorrect. This is an unborated water source that is the only source that is design class 1, but not the preferred source.

Technical References: OP AP-22

References to be provided to applicants during exam: None

Learning Objective: 40509 Discuss abnormal conditions associated with the SFP cooling system

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Past NRC Exam

No

Last Two NRC Exams

No

Question History:**Question Cognitive Level:**

Memory/Fundamental

Comprehensive/Analysis

X

10CFR Part 55 Content:

55.41.10

Difficulty: 2.0

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	2
K/A #	035 K5.01
Rating	3.4

035 K5.01 Knowledge of operational implications of the following concepts as they apply to the S/GS: Effect of secondary parameters, pressure, and temperature on reactivity.

Question 33

What reactor power and core life conditions would provide the GREATEST amount of positive reactivity for a faulted Main Steam line event?

- A. Beginning of core life, hot zero power
- B. Beginning of core life, 100% power
- C. End of core life, hot zero power
- D. End of core life, 100% power

Proposed Answer: C. End of core life, hot zero power

Explanation:

KA is met because it asks to address the effect of a loss of steam generator pressure and the effect on reactivity, by way of a temperature change.

- A. Incorrect. BOL is not the worst case, but plausible if its thought that because MTC is smaller, a deeper cooldown is required which ultimately leads to more reactivity. HZP is correct because there is more steam generator inventory, which does cause more of a cooldown.
- B. Incorrect. BOL and 100% are not correct. Plausible If the inventory effects are reversed (thought that at a higher steam load, there is more inventory) and more cooldown is needed at BOL.
- C. Correct. EOL is worst case due to the magnitude of MTC and at HZP due to a greater inventory.
- D. Incorrect. EOL is worst case but inventory is greater at HZP not full power..

Technical References: LTAA6

References to be provided to applicants during exam: None

Learning Objective Apply fundamental topics associated with the Steam Generators. (40557)

Question Source:	Bank #41 DCPN NRC Exam 07/2011	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.1	

Difficulty: 2.0

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	2
K/A #	041 A4.04
Rating	2.7

041 A4.04 Ability to manually operate and/or monitor in the control room: Pressure mode

Question 34

GIVEN:

- Unit 1 is at 3% power, EOL
- Steam Dumps are in AUTOMATIC maintaining steam generator pressure

The Steam Dump pressure controller, HC-507 setpoint is changed from 83.8% to 82.5%.

RCS Tave will _____ 1) _____ and indicated steam dump demand will _____ 2) _____.

- A. 1) remain the same 2) lower
- B. 1) remain the same 2) rise
- C. 1) lower 2) rise
- D. 1) lower 2) lower

Proposed Answer: C. 1) lower 2) rise

Explanation:

At 3%, steam dumps are in the PRESSURE mode. Lowering the setpoint, causes a lowering of the RCS temperature that the steam dumps will maintain (from 547°F to 544°F). This will result in an increased steam flow (demand).

- A. Incorrect. Tave lowers. Normal plant response of steam dumps is to maintain constant Tave, also plausible if the effect of the setpoint change was thought to close dumps, steam flow (demand) could be thought to lower.
- B. Incorrect. Tave lowers. Its thought the function is to maintain constant Tave, then raising steam (demand) flow to maintain Tave.
- C. Correct. Tave will lower, this will cause a rise in steam flow (demand) to maintain the lower temperature.
- D. Incorrect. It is correct that Tave lowers but the effect is to raise steam flow not lower. Plausible that lower setpoint will lower steam flow (demand).

Technical References: LC-2B

References to be provided to applicants during exam: None

Learning Objective: Describe system interrelationships between the Steam Dump System and other plant systems. (8042)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #43 DCPN NRC Exam 02/2005	X
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5	

Difficulty: 2.0

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	2
K/A #	056 K1.03
Rating	2.6

056 K1.03 Knowledge of the physical connections and/or cause-effect relationships between the Condensate System and the following systems: MFW

Question 36

A rapid load reduction is being performed on Unit 1 in accordance OP AP-25, Rapid Load Reduction or Shutdown. All Condensate Booster Pump Sets are running.

Main Feed pump (MFP) suction pressure is 230 psig and lowering slowly.

According to OP AP-25, what can be done to attempt to raise MFP suction pressure?

NOTE:

- FCV-55 - Fdwtr Htrs Bypass
- FCV-31 - Stator & H2 Clrs Flow Cont
- FCV-230 - Cnd Polishers Bypass Vlv
- TCV-23 - Gen H2 Cold Gas Temp Cont

- A. OPEN TCV-23 and CLOSE FCV-31
- B. OPEN FCV-230 and CLOSE FCV-31
- C. OPEN FCV-55 and OPEN FCV-230
- D. OPEN TCV-23 and OPEN FCV-55

Proposed Answer: D. OPEN TCV-23 and OPEN FCV-55

Explanation:

Interrelationship is the effect of operating valves in the condensate system to address a problem in the main feedwater system.

- A. Incorrect. TCV-23 is opened. Closing FCV-31 could be thought of as a way to put more flow to the booster pumps which could then increase flow (and raise pressure) to the suction of the MFPs.
- B. Incorrect. FCV-230 could increase flow but would have the undesirable effect of bypassing the polishers.
- C. Incorrect. FCV-55 can be used, however, FCV-230 would work but has the undesirable effect of bypassing the polishers and not given as a possibility.
- D. Correct. Both FCV-55 and TCV-23 put more flow to the suction of the MFPs.

References to be provided to applicants during exam: None

Technical References: OIM C-7-1, OP AP-25

Learning Objective: Describe system interrelationships between the Condensate System and other plant systems. (8718)

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Past NRC Exam

No

Question History:

Last Two NRC Exams

No

Question Cognitive Level:

Memory/Fundamental

X

10CFR Part 55 Content:
Difficulty: 2.7

Comprehensive/Analysis
55.41.5

L162 Answer Key

Examination Outline Cross-Reference

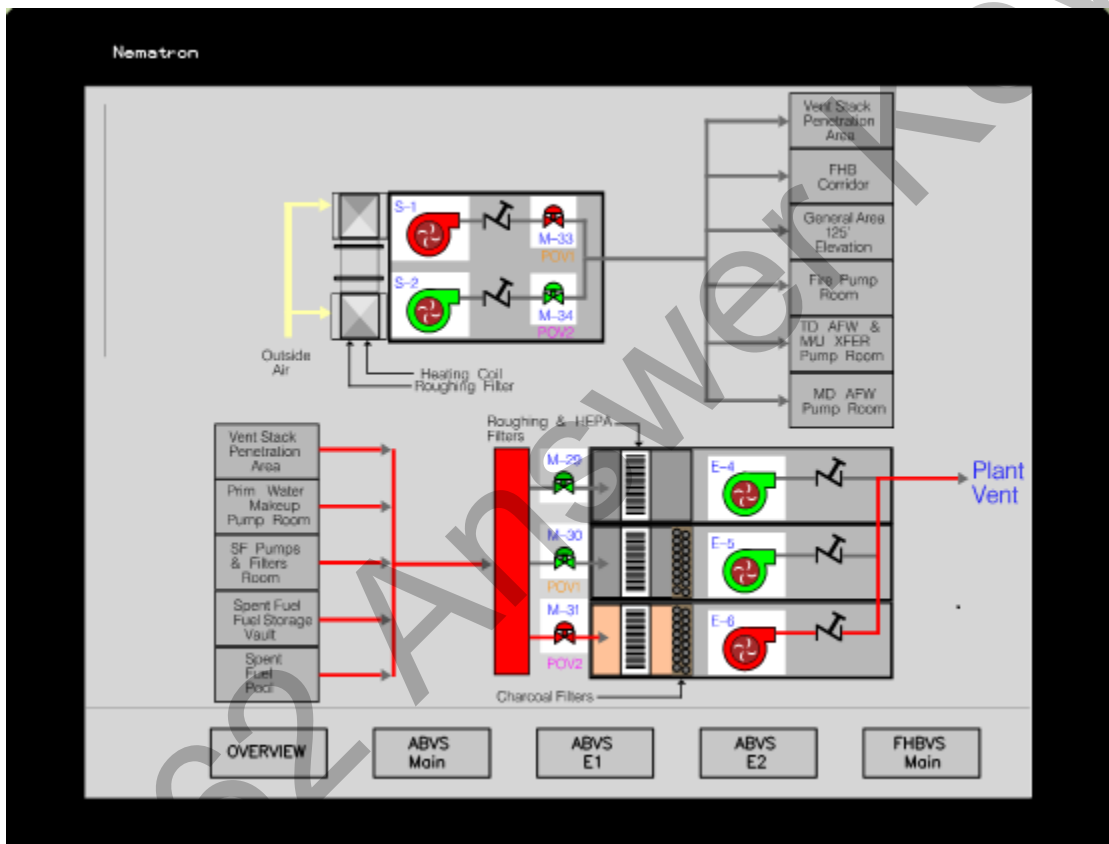
Level	RO
Tier #	2
Group #	2
K/A #	072 G2.2.44
Rating	4.2

072 G2.2.44 Area Radiation Monitoring; 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.

Question 37

PK11-10, FHB High Radiation, alarms in the Control Room. The operator confirms Spent Fuel Pool area radiation monitors, RM-58 and 59 are in HIGH alarm.

AR PK11-10 instructs the operator to "Verify that the Fuel Handling Ventilation transferred to Iodine Removal Mode". The operator notes the following on the AFHBVS monitor on VB4:



What is the status of the Fuel Handling Ventilation system?

The Fuel Handling Ventilation system is:

- A. properly aligned for Iodine removal.
- B. not properly aligned because there is a supply fan is running.
- C. not properly aligned because exhaust fan E-6 should not be running and E-4 or E-5 should be running.
- D. not properly aligned because in addition to exhaust fan E-6 running, E-5 should also be running.

Proposed Answer: A. properly aligned for Iodine removal

Explanation:

- A. Correct. The proper alignment is one supply fan and either exhaust fan E-5 or E-6. The alignment shown is correct.
- B. Incorrect. A supply fan should be running. The capacity of the exhaust fans is greater than the supply fans, so the supply fans do not need to trip to maintain a negative pressure in the FHB. Additionally, the Aux Bldg ventilation stops supply fans in the "safeguards" mode.
- C. Incorrect. E-4 is the fan that is not used for Iodine removal. Either E-5 or E-6, depending on the which is selected by a switch on VB4. Plausible because it may not be known that one of the fans isn't used but whether its E-4 or E-6 is not.
- D. Incorrect. If it's thought that ventilation that the additional exhaust fan must run to ensure the proper pressure is maintained in the FHB with the supply fan running.

Technical References: AR PK11-10, LH-7, and LH-1

References to be provided to applicants during exam: None

Learning Objective: 37547 - Describe controls, indications, and alarms associated with the FHBVS

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

Past NRC Exam

Last Two NRC Exams

X

No

No

Question History:

Question Cognitive Level:

Memory/Fundamental

Comprehensive/Analysis

X

10CFR Part 55 Content:

55.41.7

Difficulty: 2.3

Examination Outline Cross-Reference

086 K4.06 Knowledge of design feature(s) and/or interlock(s) which provide for the following: CO2

Level	RO
Tier #	2
Group #	2
K/A #	086 K4.06
Rating	3.0

Question 38

CO2 (CARDOX) can be initiated from the Control Room to the:

1. Emergency Diesel Generators
2. Cable Spreading Rooms
3. 12 kV Switchgear Rooms

- A. 1 ONLY
- B. 1 and 2 ONLY
- C. 2 and 3 ONLY
- D. 1 and 2 and 3

Proposed Answer: B. 1 and 2 ONLY

Explanation:

- A. Incorrect. EDGs is correct but the answer is not complete, CARDOX can also be initiated to the Cable Spreading Rooms.
- B. Correct. CARDOX can be actuated to the diesels, and to the cable spreading rooms.
- C. Incorrect. CARDOX to 12 kV cannot be actuated from the control room, there are hose reels for the 12 kV room.
- D. Incorrect. CARDOX to 12 kV cannot be actuated from the control room, there are hose reels for the 12 kV room.

Technical References: LK-2B, OP K-2B:II

References to be provided to applicants during exam: None

Learning Objective: Describe the operation of the Cardox System. (3707)

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

Past NRC Exam

Last Two NRC Exams

Question History:

Question Cognitive Level:

Memory/Fundamental

Comprehensive/Analysis

10CFR Part 55 Content:

Difficulty: 2.7

55.41.7

X

No

No

X

Examination Outline Cross-Reference**APE 008 G2.4.3 - Pressurizer Vapor Space Accident: Knowledge of the operational implications of EOP warnings, cautions, and notes..**

Level	RO
Tier #	1
Group #	1
K/A #	APE 008 G2.4.3
Rating	3.8

Question 39

GIVEN:

- A Pressurizer PORV is partially open.
- The crew is performing the ECCS flow reduction actions of E-1.2, Post LOCA Cooldown and Depressurization
- One Charging and both SI pumps are running
- The RHR pumps are secured
- Conditions are met for securing one of the running SI pumps

When the SI pump is secured, RCS pressure begins to decrease.

According to a Note prior to the step in E-1.2, what action should be taken?

- A. Immediately restart the SI pump.
- B. Restart an RHR pump in the SI mode.
- C. Proceed on to securing the second SI pump.
- D. Wait for RCS pressure to stabilize or increase before taking action to stop the second SI pump.

Answer: D. Wait for RCS pressure to stabilize or increase before taking action to stop the second SI pump.

Explanation:

From E-1.2 background: After an SI pump is stopped, RCS pressure may decrease to a new equilibrium value where the reduced SI flow again matches leakage from the RCS. The criteria for stopping the next SI pump has been calculated assuming steady-state conditions. Hence, to ensure that these criteria are appropriate, RCS pressure and subcooling should be allowed to stabilize or increase before additional SI pumps are stopped.

- A. Incorrect, note states pressure should be allowed to stabilize or increase. Plausible because there are conditions that require restarting ECCS pumps, such as loss of subcooling on the foldout page.
- B. Incorrect, RHR pumps are restarted if subcooling is low when checking reduction criteria or if Tave is less than 350F prior to stopping an SI pump .
- C. Incorrect, a period of time should pass to ensure RCS pressure will stabilize. Plausible if its thought this is the normal response and securing the next pump will be done if pressure stabilizes.
- D. Correct - E-1.2 NOTE - After stopping any ECCS Pp, RCS Pressure should be allowed to stabilize or increase before stopping another ECCS Pp.

Technical References: E-1.2, foldout page, and background

References to be provided to applicants during exam: None

Learning Objective: 6743 - Explain PZR response during ECCS reduction sequence

Question Source: (note changes; attach parent)	Bank #40 DCPD NRC10/2016 Modified Bank # New	X
Question History:	Past NRC Exam Last Two NRC Exams	Yes Yes
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5	

Difficulty: 2.7

L162 Answer Key

Examination Outline Cross-Reference

EPE 009 G2.4.21 – Small Break LOCA: 2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Level	RO
Tier #	1
Group #	1
K/A #	EPE 009 G2.4.21
Rating	4.0

Question 40

- 1) What size LOCA is assumed to have the greater probability to cause an inadequate core cooling condition?
- 2) What parameter is a direct indication that an inadequate core cooling condition exists?
 - A. 1) Small break 2) RVLIS level below the top of the core
 - B. 1) Small break 2) Core Exit Thermocouples greater than 1200°F
 - C. 1) Large break 2) RVLIS level below the top of the core
 - D. 1) Large break 2) Core Exit Thermocouples greater than 1200°F

Proposed Answer: B. 1) Small break 2) Core Exit Thermocouples greater than 1200°F

Explanation:

- A. Incorrect. A small break is the likely failure, however, CETCs are the indication. RVLIS is used in conjunction with CETCs to indicate challenges to core cooling but RVLIS level can be below the top of the core and not have an inadequate core cooling issue.
- B. Correct. The small break is the probable break to lead to a challenge to core cooling. CETCs greater than 1200°F indicate inadequate core cooling.
- C. Incorrect. Large breaks are plausible because of the potential for the core to become partially uncovered for a short period of time, however, accumulators will refill the core and will not cause ICC. RVLIS by itself does not indicate a challenge to core cooling
- D. Incorrect. Large breaks cause a potential for the core to become partially uncovered for a short period of time, however, accumulators will refill the core and will not cause ICC. CETCs above 1200°F indicate inadequate core cooling.

Technical References: LMCDFRC-Mitigating Core Damage-Core Cooling, page 26, 27. F-0

References to be provided to applicants during exam: None

Learning Objective:Differentiate between:

- a. Adequate Core Cooling
- b. Degraded Core Cooling
- c. Inadequate Core Cooling

Question Source:

(note changes; attach parent)

Bank	
Modified Bank	
New	X
Past NRC Exam	No
Last Two NRC Exams	No
Memory/Fundamental	X
Comprehensive/Analysis	
55.41.8	

Question History:

Question Cognitive Level:

10CFR Part 55 Content:

Difficulty: 2.0

L162 Answer Key

Examination Outline Cross-Reference

APE 015/017 AK3.07 Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): Ensuring that S/G levels are controlled properly for natural circulation enhancement.

Level	RO
Tier #	1
Group #	1
K/A #	APE
	015/017
	AK3.07
Rating	4.1

Question 41

The crew is preparing to depressurize steam generators to inject accumulators in accordance with EOP ECA-0.0, Loss of All Vital AC Power.

In accordance with the background document for ECA-0.0, during the depressurization, narrow range level in 1 intact steam generator(s) must be maintained greater than 15% to ensure that 2 .

- A. 1) ALL 2) steam generator pressure does not lower uncontrollably
- B. 1) ALL 2) sufficient heat transfer capability exists to remove heat from the RCS
- C. 1) ONE 2) steam generator pressure does not lower uncontrollably
- D. 1) ONE 2) sufficient heat transfer capability exists to remove heat from the RCS

Proposed Answer: D. 1) ONE 2) sufficient heat transfer capability exists to remove heat from the RCS

Explanation:

If a loss of all AC Vital power has occurred, the RCPs are off (loss of RC Flow) and heat removal is via natural circulation. Natural circulation is enhanced by maintaining sufficient steam generator level.

NOTE: Obj to know basis of EOP steps is RO and SRO knowledge. Also, RO expected from multiple EOPs that adequate heat sink is 15% in one steam generator.

- A. Incorrect. The step requires at least one steam generator greater than 15%. The pressure lowering uncontrollably is the basis for maintaining level above 15% in E-3, Steam Generator Tube Rupture.
- B. Incorrect. The step requires at least one steam generator greater than 15%. Reason is correct.
- C. Incorrect. Only one steam generator is required. The pressure lowering uncontrollably is the basis for maintaining level above 15% in E-3, Steam Generator Tube Rupture
- D. Correct. One steam generator is required. Per the background document, The analysis basis for ECA-0.0 requires that the level in at least one intact steam generator (SG) be above the top of the SG U-tubes to ensure that sufficient heat transfer capability exists to remove heat from the RCS via either natural circulation or reflux boiling after the RCS saturates which is accomplished by maintaining level above 15%.

Technical References: ECA-0.0 step 18, ECA-0.0 background, E-3 background, LPECA-0

References to be provided to applicants during exam: None

Learning Objective: 8905 - Explain the operator actions that can initiate or enhance natural circulation

Question Source:

(note changes; attach parent) Bank #
Modified Bank #

Question History:	New	X
Question Cognitive Level:	Past NRC Exam	No
	Last Two NRC Exams	No
	Memory/Fundamental	X
10CFR Part 55 Content:	Comprehensive/Analysis	
Difficulty: 2.7	55.41.5	

PRA initiator to CDF – loss of offsite power

L162 Answer Key

Examination Outline Cross-Reference

APE 022 AA1.08 Ability to operate and / or monitor the following as they apply to the Loss of Reactor Coolant Makeup: VCT level

Level	RO
Tier #	1
Group #	1
K/A #	APE 022 AA1.08
Rating	3.4

Question 42

Unit 1 is at 100% power.

The operator stops the running charging pump when it was showing signs of cavitation. The operator reports VCT outlet valves 112B and 112C are open and RWST outlet valves 8805A and 8805B are closed.

When checking VCT level, which of the following VCT level channel failures would be expected to have caused the observed indications?

- A. LT-112 failed LOW
- B. LT-112 failed HIGH
- C. LT-114 failed LOW
- D. LT-114 failed HIGH

Proposed Answer: B. LT-112 failed HIGH

Explanation:

Question addresses a "loss of makeup" (charging) from the perspective of the VCT makeup failure that leads to a loss of charging by causing VCT to drain and there is no swapover to RWST. Ultimately, there could be a loss of all charging due to gas intrusion from the VCT to the suction of the charging pumps.

- A. Incorrect. Plausible if failure is misconstrued. LT-112 LOW would initiate continuous makeup and the VCT fills and diverts. Charging and letdown unaffected.
- B. Correct. Auto makeup is lost. VCT level will decrease. No auto swapover to the RWST, at 5% will occur (requires both channels, LT112 and LT114). VCT empties, charging flow decreases to 0 gpm. Pressurizer level decreases, letdown isolates and heaters de-energize.
- C. Incorrect. VCT level would be maintained by LT112. If level on LT-112 were to also lower to 5%, auto swapover to RWST. Charging flow would be maintained. Plausible if the functions for LT112 are assumed for LT114.
- D. Incorrect. This fully opens LCV-112A. VCT level decreases to 14%, auto makeup from LT-112, makeup is greater than letdown. Plausible if its thought auto makeup is defeated (as for LT112).

Technical References: OIM B-1-4. OP AP-19 appendix A

References to be provided to applicants during exam: None

Learning Objective: 40449 - Discuss abnormal conditions associated with the CVCS

Question Source:

(note changes; attach parent)

Bank	
Modified Bank	
New	X
Past NRC Exam	No
Last Two NRC Exams	No

Question History:

Question Cognitive Level:

Memory/Fundamental
Comprehensive/Analysis

X

10CFR Part 55 Content:

55.41.6

Difficulty: 2.3

L162 Answer Key

Examination Outline Cross-Reference

APE 025 AK1.01 Knowledge of the operational implications of the following concepts as they apply to Loss of Residual Heat Removal System: Loss of RHR during all modes of operation

Level	RO
Tier #	1
Group #	1
K/A #	APE 025 AK1.01
Rating	3.9

Question 43

GIVEN:

- Unit 1 is in MODE 4
- MSIVs are closed
- One RHR pump is out of service

The running RHR pump trips and cannot be restarted.

Which of the following is the preferred method of RCS temperature control in accordance with OP AP-16, Malfunction of the RHR System?

- A. Fill and spill to the containment floor
- B. Charging injection to the Cold Legs
- C. Dump steam to the Atmosphere
- D. Feed and bleed

Proposed Answer: C. Dump steam to the Atmosphere

Explanation:

- A. Incorrect. Plausible because this is a MODE 5 or 6 option if unable to perform feed and bleed.
- B. Incorrect. Plausible because this is a MODE 5 or 6 option if steam generators are not available.
- C. Correct. Step 2 of OP AP-16 states: Control RCS temperature using condenser steam dumps as necessary or Control RCS temperature by dumping steam to the atmosphere using S/G 10% atmospheric steam dumps. The condenser is not available because the MSIVs are closed. The action is dump steam using the 10% steam dumps to atmosphere.
- D. Incorrect. Feed and bleed is a form of heat removal done when a loss of RHR occurs in modes 5 and 6, not in AP-16.

Technical References: OP AP-16, OP AP SD-0 appendix E and F

References to be provided to applicants during exam: None

Learning Objective:

Question Source: (note changes; attach parent)	Bank #44 DCPN NRC Exam 08/2014 Modified Bank # New Past NRC Exam	X Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 X
10CFR Part 55 Content: Difficulty: 2.3	55.41.10	

L162 Answer Key

Examination Outline Cross-Reference

Level	RO
Tier #	1
Group #	1
K/A #	APE 026 AA2.04
Rating	2.5

APE 026 AA2.04 Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The normal values and upper limits for the temperatures of the components cooled by CCW

Question 44

GIVEN:

- Unit 1 is at 100% power
- Thermal Barrier CCW outlet valve, FCV-357, closes on high flow
- The crew enters OP AP-11, Malfunction of the Component Cooling Water System, section B, CCW System Inleakage
- RCP radial bearing temperature is 180°F and rising at 5°F/minute.

For the current plant conditions, in accordance with OP AP-11, tripping the reactor and tripping of the RCP will FIRST be required in _____ minutes.

- A. the next 5
- B. 8
- C. 9
- D. 11

Proposed Answer: C. 9.

Explanation:

- A. Incorrect. This would be correct if there was also a loss of seal injection flow. No trip is required unless both occur.
- B. Incorrect. Temperature in 8 minutes will be 220°F, less than the setpoint of 225°F.
- C. Correct. Radial bearing temperature limit is 225F and will be reached in 9 minutes at the current rate.
- D. Incorrect. Temperature will be 235°F, above the setpoint. Also corresponds to seal water outlet temperature setpoint.

Technical References: OP AP-11 section B**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:**

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Past NRC Exam

No

Question History:

Last Two NRC Exams

No

Question Cognitive Level:

Memory/Fundamental

Comprehensive/Analysis

X

10CFR Part 55 Content:

55.41.7

Difficulty: 3.0

Examination Outline Cross-Reference

APE 027 AK2.03 Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners

Level	RO
Tier #	1
Group #	1
K/A #	APE 027 AK2.03
Rating	2.6

Question 45

The unit is operating at 50% power.

PZR Press Control, HC-455K, output rises to 100%.

Pressurizer pressure will:

- A. rise until a PORV actuates and then pressure will cycle between the close interlock and PORV setpoint.
- B. rise to the reactor trip setpoint.
- C. lower until the Back-up heaters energize and then stabilize.
- D. lower to the reactor trip setpoint.

Proposed Answer: D. lower to the reactor trip setpoint.

Explanation:

- A. Incorrect. A PORV will open (and close at 2135 psig), but the spray valves open (and heaters turn off and pressure will lower.
- B. Incorrect. Full output will cause the spray valves and a PORV open, this will lower and not raise pressure to the trip setpoint.
- C. Incorrect. Pressure will lower but the master controller at 100% will prevent energizing the heaters.
- D. Correct. The spray valves and a PORV will open to lower pressure. The heaters will not energize, as a result pressure continues to lower and reactor trip will occur on low pressure.

Technical References: OIM A-4-4A

References to be provided to applicants during exam: None

Learning Objective: 36926 Discuss abnormal conditions associated with the Pzr, Pzr Pressure and Level Control System

Question Source:	Bank #A-1058 DCPD Bank	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	
Difficulty: 3.0		

Examination Outline Cross-Reference

EPE 029 EA1.12 Ability to operate and monitor the following as they apply to a ATWS: M/G set power supply and reactor trip breakers

Level	RO
Tier #	1
Group #	1
K/A #	EPE 029 EA1.12
Rating	4.1

Question 46

The Unit 1 is at 100% power when the operator takes the Reactor Trip Switch to "TRIP".

If, as a minimum, 1) RED Reactor Trip Breaker lights remain lit, the operator must de-energize 2) to cause the rods to insert.

- A. 1) EITHER 2) 13D OR 13E
- B. 1) EITHER 2) 13D AND 13E
- C. 1) BOTH 2) 13D OR 13E
- D. 1) BOTH 2) 13D AND 13E

Proposed Answer: D. 1) BOTH 2) 13D AND 13E

Explanation:

- A. Incorrect. If one reactor trip breaker opens, then 13D and E do not need to be de-energized to cause the rods to insert.
- B. Incorrect. If one reactor trip breaker opens, then 13D and E do not need to be de-energized to cause the rods to insert. The action would be to open both 13D and 13E breaker.
- C. Incorrect. Both red lights lit would require the operator to de-energize the rod drive MG sets by opening BOTH (not one) 13D and E breakers.
- D. Correct. Both red lights indicate both trip breakers are closed. BOTH supply breakers to the MG sets must be opened (they are in parallel) to de-energize the rods and cause them to insert.

Technical References: OIM A-3-1

References to be provided to applicants during exam: None

Learning Objective: 40752 Describe the basic flow path of the Rod Control System

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.6	

Difficulty: 2.3

Examination Outline Cross-Reference

EPE 038 EK1.01 Knowledge of the operational implications of the following concepts as they apply to the SGTR: Use of steam tables

Level	RO
Tier #	1
Group #	1
K/A #	EPE 038 EK1.01
Rating	3.1

Question 47

GIVEN:

- A steam generator tube rupture has occurred on Unit 2
- A loss of offsite power has occurred
- The crew is completing a cooldown to 510°F in accordance with EOP E-3, Steam Generator Tube Rupture

The operator has been instructed to determine a new setpoint for the steam dump valves.

Approximately, what should be the new setting?

- A. 73.0% on 40% Steam Dump Valve Pressure controller, HC-507
- B. 73.0% on the intact steam generator 10% steam dump controllers
- C. 61.0% on 40% Steam Dump Valve Pressure controller, HC-507
- D. 61.0% on the intact steam generator 10% steam dump controllers

Proposed Answer: D. 61.0% on the intact steam generator 10% steam dump controllers

Explanation:

- A. Incorrect. A loss of offsite power means the cooldown will be done using the 10% steam dumps because there are no circ water pumps running so the condenser is not available. Also the new setpoint is a conversion of steam generator pressure when the cooldown is completed, approximately 730 psig (744.5 psia), divided by 12 psig. 73.0 would be the setpoint of converting 730 to a percent (or based on a scale of 0 to 1000).
- B. Incorrect. The 10% steam dumps will be used, but the setpoint will be closer to 61%
- C. Incorrect. Setpoint is correct, but the cooldown will be done with 10% steam dumps.
- D. Correct. 510°F equals a Psat of approximately 730 psig (744.5 psia). The setpoint is $730/12 =$ approximately 61% (60.8%)

Technical References: steam tables, E-3

References to be provided to applicants during exam: steam tables

Learning Objective: 77975 Given a set of Steam Tables, determine the: saturation temperature, saturation pressure,

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

Past NRC Exam

Last Two NRC Exams

X

No

No

Question History:

Question Cognitive Level:

Memory/Fundamental

Comprehensive/Analysis

X

10CFR Part 55 Content:

55.41.14

Difficulty: 3.0

Examination Outline Cross-Reference

APE 040 AK2.02 Knowledge of the interrelations between the Steam Line Rupture and the following: Sensors and detectors

Level	RO
Tier #	1
Group #	1
K/A #	APE 040 AK2.02
Rating	2.6

Question 48

A steamline break occurs in Containment.

A Safety Injection signal will occur when the minimum coincidence of 1) Containment pressure channels reach the actuation setpoint.

Adverse containment values will be used when containment pressure reaches a minimum of 2).

- A. 1) 2 of 3 2) 3 psig
- B. 1) 2 of 3 2) 22 psig
- C. 1) 2 of 4 2) 3 psig
- D. 1) 2 of 4 2) 22 psig

Proposed Answer: A. 1) 2 of 3 2) 3 psig

Explanation:

- A. Correct. Unlike Pressurizer low pressure, containment high pressure is 2 of 3 detectors. Adverse containment pressure setpoint is 3 psig.
- B. Incorrect. 22 psig is the high-high containment spray setpoint.
- C. Incorrect. Coincidence is 2 of 3. The coincidence for Containment Spray and Phase B is 2 of 4 channels (vice only 3 for SI)
- D. Incorrect. This is the correct setpoint for Phase B and Containment Spray.

Technical References: OIM B-6-5, B-6-8

References to be provided to applicants during exam: None

Learning Objective: 37048 Analyze automatic features and interlocks associated with the RPS

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #48 DCPN NRC Exam 01/2010	X
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Difficulty: 2.3

Examination Outline Cross-Reference

Level	RO
Tier #	1
Group #	1
K/A #	APE 054 AK3.04
Rating	4.4

APE 054 AK3.04 Knowledge of the reasons for the following responses as they apply to the Loss of Main Feedwater (MFW): Actions contained in EOPs for loss of MFW

Question 49

Unit 1 is at 100% power when 11 Main Feedwater pump trips and the crew enters OP AP-15, Loss of Feedwater Flow.

A programmed ramp has initiated.

In accordance with a Caution in OP AP-15, why is it important to prevent excessive Tav_g/Tref deviation causing the Group II Steam Dump valves to open?

- A. Resulting swell could cause P-14 actuation.
- B. Resulting shrink could cause a reactor trip on low steam generator level.
- C. The excessive demand could cause a reactor trip on low pressurizer pressure.
- D. The excessive demand could cause rods to drive below the Rod Insertion Limit.

Proposed Answer: A. Resulting swell could cause P-14 actuation.

Explanation:

- A. Correct. OP AP-15 cautions: If excessive demand causes the Group II steam dumps to open, S/G swell will likely cause a P 14 actuation
- B. Incorrect. Steam generators will swell, not shrink. Shrink is the reason for starting the AFW pumps at step 1.
- C. Incorrect. Steam dumps remove heat and will drive Tav_g lower. RCS pressure lowers as the colder RCS water is cooled. Plausible because excessive cooldown can cause SI or low pressure trip.
- D. Incorrect. Plausible to think that maintaining a large Tav_g/Tref difference will drive rods deeper and potentially below the RIL.

Technical References: OP AP-15, LPA-15

References to be provided to applicants during exam: None

Learning Objective: Given an abnormal condition, summarize the major actions of OP AP-15 to mitigate an event in progress. (3477Q)

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Past NRC Exam

No

Last Two NRC Exams

No

Question History:**Question Cognitive Level:**

Memory/Fundamental

X

Comprehensive/Analysis

10CFR Part 55 Content:

55.41.5

Difficulty: 2.0

Examination Outline Cross-Reference

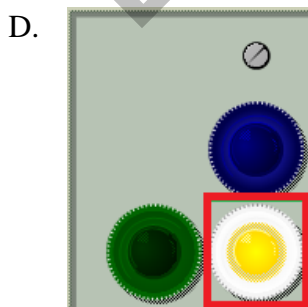
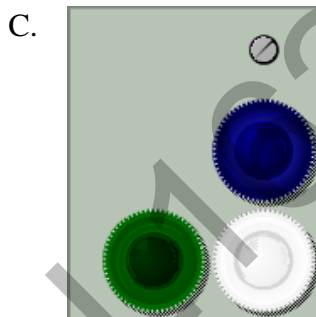
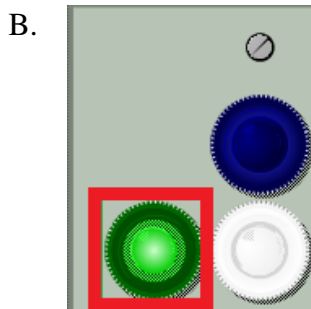
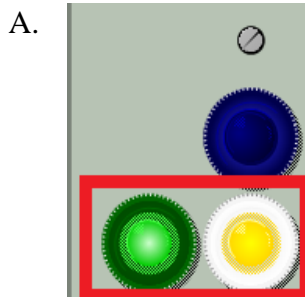
Level	RO
Tier #	1
Group #	1
K/A #	APE056 AA2.45
Rating	3.6

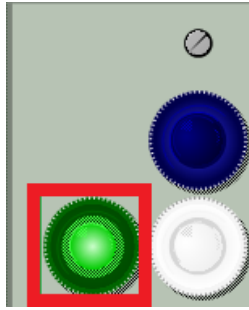
APE 056 AA2.45 Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Indicators to assess status of ESF breakers (tripped/not-tripped) and validity of alarms (false/not-false)

Question 50

A loss of offsite power occurs on Unit 2 and subsequently PK18-17, 4KV BUS F BUS OR SU FDR UV, is lit.

What Green and White light indications for a pump powered by 4 kV Bus F would confirm the alarm that a loss of the bus has occurred? (ONLY lights outlined in RED are lit)





Proposed Answer: B.

Explanation:

4 KV bus F powers ESF pumps – ASW pump 21, AFW pump 23, CCP, CCW and SI pumps 21.

- A. Incorrect. Loss of control power makes white light go out. Plausible if its thought the lights are powered by control power and would be on with or without bus power.
- B. Correct. The white light is indication the bus is energized, the red and green lights are powered by DC control power. Therefore, if the bus is de-energized, the green light will be on but the white light will be out.
- C. Incorrect. If the bus powered all the lights, this would be correct.
- D. Incorrect. This is backwards of the expected indication and if its thought the white light indicates control power.

Technical References: OIM J-1-1, OP AP-27

References to be provided to applicants during exam: None

Learning Objective: 41081 Discuss abnormal conditions associated with the 4KV System

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

Past NRC Exam

Last Two NRC Exams

X

No

No

Question History:

Question Cognitive Level:

Memory/Fundamental

Comprehensive/Analysis

X

10CFR Part 55 Content:

55.41.8

Difficulty: 2.3

Examination Outline Cross-Reference

APE 057 AA1.06 Ability to operate and / or monitor the following as they apply to the Loss of Vital AC Instrument Bus: Manual control of components for which automatic control is lost

Level	RO
Tier #	1
Group #	1
K/A #	APE 057 AA1.06
Rating	3.5

Question 51

GIVEN:

- Unit 2 tripped from full power
- All steam generator narrow range levels are now 60% and rising
- AFW flow hand controllers are in AUTO with 30% demand

A loss of PY-23 occurs. As a result, power is lost to the hand controllers for Steam Generator AFW Control valves, LCV-110 and LCV-111.

When power is restored to PY-23 the Control Room hand controllers for LCV-110 and LCV-111 will be in:

- A. MANUAL with 30% demand
- B. MANUAL with 100% demand
- C. AUTO with 30% demand
- D. AUTO with 100% demand

Proposed Answer: B. MANUAL with 100% demand

Explanation:

- A. Incorrect. HC will be in MANUAL, however, the valves go to the full open position, 100%. 30% is plausible as it could be the fail position is "as is".
- B. Correct. Upon restoration of power, HC will be in MANUAL with demand output of 100% (4.0 mA) and valve fully open
- C. Incorrect. At the Hot Shutdown Panel, If HIC (controller) is in AUTO when power is lost, HIC will fail as-is and demand signal will continue to pass through to the valve. Upon restoration of power, HIC will be in AUTO.
- D. Incorrect. Upon loss and restoration of power to the Tricon, setpoint will return to the value that existed before power was lost. Tricon power was not lost, power to the controllers was lost.

Technical References: OP O-2 attachment 1**References to be provided to applicants during exam:** None**Learning Objective:** 37637 - Analyze automatic features and interlocks associated with the AFW system

Question Source:	Bank #51 DCPN NRC Exam 08/2014	X
(note changes; attach parent)	Modified Bank # New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Difficulty: 3.0

L162 Answer Key

Examination Outline Cross-Reference

APE 058 G2.2.37 Loss of DC Power: Ability to determine operability and/or availability of safety related equipment.

Level	RO
Tier #	1
Group #	1
K/A #	APE 058 G2.2.37
Rating	3.6

Question 52

Unit 1 is at 100% power.

A loss of DC bus 12 and PY-12 occurs.

Per OP AP-23, Loss of Vital DC Bus, which of the following occurs?

- 1) Main Feedwater is isolated
- 2) No control for one Emergency Diesel Generator from the Control Room
- 3) One Train of SSPS Slave relays is de-energized

- A. 1 and 2
- B. 1 and 3
- C. 2 ONLY
- D. 3 ONLY

Proposed Answer: A. 1 and 2

Explanation:

- A. Correct. Loss of DC bus 12 cause the main reg valve solenoids to de-energize, isolating main Feedwater. The SSPS relays are powered from PY-11 and 14 and are not affected. Control of one diesel (1-2) is affected due to the loss of control power.
- B. Incorrect. The SSPS relays are powered from PY-11 and PY-14.Plausible the power supplies would be PY-11 and PY-12.
- C. Incorrect. There is a loss of control for one diesel but there is also a loss of power to the MFW solenoids and MFW isolates.
- D. Incorrect. Some SSPS solenoids do lose power, however, only for PY-11 or PY-14, not PY-12.

Technical References: OP AP-23

References to be provided to applicants during exam: None

Learning Objective:State the power supplies to Pressurizer, Pressure & Level Control System components. (9990)

Question Source:

(note changes; attach parent)

Bank #	
Modified Bank #	
New	X
Past NRC Exam	No
Last Two NRC Exams	No
Memory/Fundamental	X
Comprehensive/Analysis	

Question History:

Question Cognitive Level:

10CFR Part 55 Content:

Difficulty: 3.3

55.41.7

PRA – Vital 125 VDC is number 1 system in risk importance

L162 Answer Key

Examination Outline Cross-Reference

Level	RO
Tier #	1
Group #	1
K/A #	APE 062 AK3.03
Rating	4.0

APE 062 AK3.03 Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: Guidance actions contained in EOP for Loss of nuclear service water

Question 53

The crew is aligning Unit 1 for Cold Leg Recirculation in accordance with E-1.3, Transfer to Cold Leg Recirculation.

Only one train of ASW is available.

When the crew completes the alignment, the reason only one RHR heat exchanger and three CFCUs will be in operation is to prevent:

- A. runout of the CCW pump
- B. runout of the ASW pump
- C. flashing and water hammer in the ASW system
- D. exceeding CCW system temperature design limit

Proposed Answer: D. exceeding CCW system temperature design limit

Explanation:

Design of the ASW (Diablo equivalent to Nuclear Service Water) system is to adequately remove CCW heat. EOP E-1.3 includes operator actions to limit the heat loads during post-LOCA cold-leg recirculation if less than two ASW pumps and two CCW heat exchangers are in Service (a loss of one train).

- A. Incorrect. With a single train of CCW in service, its plausible the student could focus on the single CCW train and believe runout is a possibility. The problem is due to the lack of cooling (from ASW), CCW heat removal is reduced to the point that if loads are not restricted, design temperature could be exceeded.
- B. Incorrect. The system is designed for one pump to supply both CCW trains during normal operation without runout. Plausible because, if one train is supplying both trains of CCW, then its possible to think the loads could cause the only running ASW pump to approach runout.
- C. Incorrect. Temperature remains less than saturation. Plausible because if two trains of CCW were put into service with only one train of ASW, temperature will rise.
- D. Correct. The heat load of all the loads on the CCW with only one ASW train could cause the system to not be able to meet its purpose to remove heat from the CCW system and the CCW system could exceed its design temperature.

Technical References: E-1.3, LF-2

References to be provided to applicants during exam: None

Learning Objective: 8105 - Explain significant CCW system design features and the importance to nuclear safety

Question Source:	Bank #14 DCPN NRC (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	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.8	
Difficulty: 2.7		

L162 Answer Key

Examination Outline Cross-Reference

Level	RO
Tier #	1
Group #	1
K/A #	APE 065 AA2.08
Rating	2.9

APE 065 AA2.08 Ability to determine and interpret the following as they apply to the Loss of Instrument Air: Failure modes of air-operated equipment.

Question 54

Unit 1 is tripped due to a loss of instrument air.

Backup nitrogen is not available to a 10% steam dump valve.

- 1) Without instrument air or nitrogen, the 10% steam dump valve fails:
- 2) The operator can regain control of the valve by:
 - A. 1) open
2) placing the AUTO/MANUAL controller in MANUAL and using the increase/decrease pushbuttons on VB3
 - B. 1) open
2) using the Cut-in toggle switch and the OPEN/CLOSE switch on VB3
 - C. 1) closed
2) placing the AUTO/MANUAL controller in MANUAL and using the increase/decrease pushbuttons on VB3
 - D. 1) closed
2) using the Cut-in toggle switch and the OPEN/CLOSE switch on VB3

Proposed Answer: D. 1) closed 2) using the Cut-in toggle switch and the OPEN/CLOSE switch on VB3

Explanation:

- A. Incorrect. The valve fails closed. Plausible to think the position is open to because its function is to cooldown the plant and it has an isolation valve which could be closed if necessary. The controller is not part of the circuit when using backup air.
- B. Incorrect. The valve fails closed. The cut-in switch allows operation using backup air via the open/close switch.
- C. Incorrect. The valve fails closed, but air to operate the valve is not thru the controller.
- D. Correct. the valve fails closed. Backup air is supplied thru the cut-in switch and then controlled using the open/close switch on VB3.

Technical References: LC-2B pages 29 to 31

References to be provided to applicants during exam: None

Learning Objective: Describe controls, indications, and alarms associated with the Steam Dump System. (37810)

Question Source:	Bank #26 DCPN NRC 04/2016	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	Yes
Question Cognitive Level:	Memory/Fundamental	

10CFR Part 55 Content:
Difficulty: 2.3

Comprehensive/Analysis
55.41.7

X

L162 Answer Key

Examination Outline Cross-Reference

E11 EK1.3 Knowledge of the operational implications of the following concepts as they apply to the (Loss of Emergency Coolant Recirculation) Annunciators and conditions indicating signals, and remedial actions associated with the (Loss of Emergency Coolant Recirculation).

Level	RO
Tier #	1
Group #	1
K/A #	E11 EK1.3
Rating	3.6

Question 55

The crew is performing the actions of ECA-1.2, LOCA Outside Containment.

- 1) The crew will transition from ECA-1.2 to ECA-1.1, Loss of Emergency Coolant Recirculation, if _____ is NOT increasing when ECA-1.2 is completed.
 - 2) A major action of ECA-1.1, is to _____.
- A. 1) RCS pressure 2) continue attempts to identify and isolate the LOCA
- B. 1) Pressurizer level 2) continue attempts to identify and isolate the LOCA
- C. 1) RCS pressure 2) conserve RWST inventory
- D. 1) Pressurizer level 2) conserve RWST inventory

Proposed Answer: C. 1) RCS pressure 2) conserve RWST inventory.

Explanation: Answer:

Question addresses the condition for entry into ECA-1.1 and the remedial action to take IAW ECA-1.1. Major actions of EOPs and exit conditions are identified in the lesson material as RO knowledge.

The following are the major actions for ECA-1.1:

- Continue attempts to restore emergency coolant recirculation
- **Raise/conserv e RWST level**
- Initiate cooldown to cold shutdown
- Depressurize the RCS to minimize subcooling
- Try to add make-up to the RCS from alternate sources,
- Depressurize the S/Gs to cooldown and depressurize the RCS
- Maintain RCS heat removal

- A. Incorrect. RCS pressure is the parameter checked. ECA-1.1 does not attempt to isolate a LOCA, regardless of location. Focus is to conserve RWST and attempt to restore emergency coolant recirc.
- B. Incorrect. RCS pressure is the parameter checked. ECA-1.1 does not attempt to isolate a LOCA, regardless of location. Focus is to conserve RWST and attempt to restore emergency coolant recirc. Pressurizer level is plausible because it is a parameter that should be restored but pressure is the more immediate and therefore, indication checked.
- C. Correct. RCS pressure should respond by rising if the leak isolation is successful, otherwise, the transition to ECA-1.1 is made to take actions that include refilling and conserving RWST.
- D. Incorrect. Action is correct, but the parameter is RCS pressure, not pressurizer level.

Technical References: ECA-1.2, ECA-1.1

References to be provided to applicants during exam: None

Learning Objective: 42461 -Explain basis of emergency steps of ECA-1.1

Explain the general purpose/function (including major action categories [MACs]) of EOPs (E-1.3/4, ECA-1 series). (3551F/J)

Identify exit conditions for the EOPs. (5433)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	
Difficulty: 2.7		

L162 Answer Key

Examination Outline Cross-Reference

APE 077 AK2.07 Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Turbine / generator control.

Level	RO
Tier #	1
Group #	1
K/A #	APE 077 AK2.07
Rating	3.6

Question 56

A grid disturbance causes Triconex DEHC frequency to rise, causing speed to rise.

Once speed rises to its overspeed setpoint of _____ 1) _____, the _____ 2) _____ valves close.

- A. 1) 101% 2) reheat stop
- B. 1) 101% 2) turbine governor and reheat intercept
- C. 1) 103% 2) reheat stop
- D. 1) 103% 2) turbine governor and reheat intercept

Proposed Answer: D. 1) 103% 2) turbine governor and reheat intercept

Explanation:

The primary feature of the OPC is the 103% overspeed protection. At 103 percent, the governor and intercept valves are closed until the speed decays to approximately 101 percent

- A. Incorrect. Reheat stop valves do not close. Plausible because reheat valves close, but it's the intercept valves.
- B. Incorrect. Plausible because these are valves that do close, however, the setpoint is 103% to close (101% to reopen).
- C. Incorrect. The reheat stop valves do not close. Plausible because the reheat intercept valve close .
- D. Correct. Both the governor and reheat intercept valves close to slow down the turbine.

Technical References: LC-3B, pages 26, 27 and 28

References to be provided to applicants during exam: None

Learning Objective: Analyze automatic features and interlocks associated with the Turbine Control Oil System. (37644)

Question Source:

(note changes; attach parent)

Bank #	
Modified Bank #16 DCPN NRC 10/2016	X
New	
Past NRC Exam	Yes
Last Two NRC Exams (modified question)	No

Question History:

Question Cognitive Level:

Memory/Fundamental	X
Comprehensive/Analysis	

10CFR Part 55 Content:

55.41.7

Difficulty: 2.0

Examination Outline Cross-Reference

APE 001 AK1.18 Knowledge of the operational implications of the following concepts as they apply to Continuous Rod Withdrawal: Fuel temperature coefficient.

Level	RO
Tier #	1
Group #	2
K/A #	APE001 AK1.18
Rating	3.4

Question 57

A continuous rod withdrawal accident occurs at Hot Zero Power.

According to the FSAR,

- 1) Uncorrected, power will be turned by:
- 2) The reactor trip expected to occur is:
 - A. 1) Moderator Temperature Coefficient
2) Power Range High Flux - Low
 - B. 1) Moderator Temperature Coefficient
2) OTΔT
 - C. 1) Doppler Coefficient
2) Power Range High Flux - Low
 - D. 1) Doppler Coefficient
2) OTΔT

Proposed Answer: C. 1) Doppler Coefficient 2) Power Range High Flux - Low

Explanation:

- A. Incorrect. MTC does not react as quickly at Doppler, which once above the POAH, rises as quickly as fuel temperature rises. The assumed trip is PR high flux, low.
- B. Incorrect. MTC is not a factor in turning power. The OTΔT trip is the assumed trip at power.
- C. Correct. Doppler is the coefficient to turn power. Once in the power range, the reactor is assumed to trip at the PR high flux low setpoint.
- D. Incorrect. Doppler is the correct coefficient, however, OTΔT is the assumed trip for this accident at power.

Technical References: LTAA 5, pages 7 - 10

References to be provided to applicants during exam: None

Learning Objective: 3444 Explain the plant's response to reactivity addition accidents described in the FSAR.

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

Past NRC Exam

Last Two NRC Exams

Memory/Fundamental

Comprehensive/Analysis

55.41.1

X

No

No

X

Question History:

Question Cognitive Level:

10CFR Part 55 Content:

Difficulty: 3.7

Examination Outline Cross-Reference

APE 005 AK2.02 Knowledge of the interrelations between the Inoperable / Stuck Control Rod and the following: Breakers, relays, disconnects, and control room switches

Level	RO
Tier #	1
Group #	2
K/A #	APE 005 AK2.02
Rating	2.5

Question 58

During a down power, a Control Bank D rod became misaligned.

- 1) What could have caused the rod to become misaligned?
- 2) In accordance with OP AP-12B, Control Rod Misalignment, if the crew attempts to realign the rod, the relay disconnect(s) for _____ will be opened to align the rod and the bank.
 - A. 1) power cabinet urgent failure
2) only the affected rod
 - B. 1) power cabinet urgent failure
2) all unaffected rods in the bank
 - C. 1) blown lift coil fuse
2) only the affected rod
 - D. 1) blown lift coil fuse
2) all unaffected rods in the bank

Proposed Answer: D. 1) blown lift coil fuse 2) all unaffected rods in the bank

Explanation:

- A. Incorrect. Power cabinet failure would have stopped all rods in the bank from moving. The rod is aligned to the bank, therefore, its lift coil is left connected, the rest of the lift coils are opened. Plausible because the power cabinet failure would stop rod motion and if its thought the bank is aligned to the rod, not vice versa.
- B. Incorrect. Action is correct, but a power cabinet urgent failure will stop all rods.
- C. Incorrect. A blown fuse will not generate an urgent failure because the bank is moving, but will stop an individual rod from moving. Action is not to align bank to the rod but vice versa.
- D. Correct. The lift coil fuse failure would cause the rod to stop moving. The rod does not drop because the stationary or movable gripper is always energized. The action is to align rod to bank, so the unaffected rods are disconnected and only the affected rod is left connected to move to the bank.

Technical References: OP AP-12B, LA-3A page 11 and 69

References to be provided to applicants during exam: None

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

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

Past NRC Exam

Last Two NRC Exams

X

No

No

Question History:

Question Cognitive Level:

Memory/Fundamental
Comprehensive/Analysis

X

10CFR Part 55 Content:

55.41.2

Difficulty: 2.3

L162 Answer Key

Examination Outline Cross-Reference

APE 032 AK3.01 Knowledge of the reasons for the following responses as they apply to the Loss of Source Range Nuclear Instrumentation: Startup termination on source-range loss

Level	RO
Tier #	1
Group #	2
K/A #	APE032 AK3.01
Rating	3.2

Question 59

The reactor is critical in the source range below P-6, IR Permissive.

Excure Source Range channel, N31, instrument power fuse blows.

The startup will be:

- A. terminated because the reactor automatically trips.
- B. suspended and the reactor maintained critical in the source range because P-6 will not energize with only one source range available.
- C. terminated and all control rods driven in because two OPERABLE source range channels are required when the reactor is critical below P-6.
- D. suspended and the reactor maintained critical in the source range because two OPERABLE source range channels are required to raise power to P-6.

Proposed Answer: A. terminated because the reactor automatically trips.

Explanation:

- A. Correct. Loss of the instrument power fuse causes the SR High Level trip bistable to cause an automatic reactor trip.
- B. Incorrect. P-6 requires IR channels to reach the setpoint. There is no setpoint for the source range and P-6. Plausible because during the startup, the source ranges are monitored for 1 decade of overlap with the intermediate ranges and P-6 alarming, before reaching the source range trip setpoint. NOTE: all "remain critical" distractors are plausible as the trip would not occur if the trip/bypass switch was in "bypass" when the instrument fuse failed.
- C. Incorrect. Two source ranges are required below P-6 and for instances, such as critical below the RIL, rods are driven in, the reason for termination is a reactor trip.
- D. Incorrect. If the reactor trip is not recognized, maintaining stable conditions is plausible.

Technical References: OIM B-4-2, B-6-2

References to be provided to applicants during exam: None

Learning Objective: Discuss abnormal conditions associated with the NIS5992

Question Source:

(note changes; attach parent)

Bank #	
Modified Bank #	
New	X
Past NRC Exam	No
Last Two NRC Exams	No

Question History:**Question Cognitive Level:**

Memory/Fundamental	
Comprehensive/Analysis	X

10CFR Part 55 Content:

Difficulty: 2.3

55.41.2

Examination Outline Cross-Reference**APE 051 – Loss of Condenser Vacuum: G2.1.32 Ability to explain and apply system limits and precautions.**

Level	RO
Tier #	1
Group #	2
K/A #	APE 051 G2.1.32
Rating	4.0

Question 60

GIVEN:

- Unit 1 is currently at 47% power and reducing power to 40% in accordance with OP AP-7, Degraded Condenser
- PK10-11, Condenser Pressure/Level, is in alarm
- Indicated condenser pressure on VB3 is 7.8” Hg and stabilizing
- No automatic actions have occurred

In accordance with OP AP-7, with these conditions, the crew will: _____.

- A. manually trip the turbine
- B. manually trip the reactor
- C. stabilize power
- D. continue the power reduction

Proposed Answer: A. manually trip the turbine

Explanation:

The candidate must explain the reason for the action to be taken, when applying the limit for condenser pressure at 47% of 7.2". Proper procedural action is to trip the turbine below P-9 (50%).

- A. Correct. At or above 50%, the reactor is tripped (which generates a turbine trip). However, below 50%, (less than P-9, the action is to trip the turbine).
- B. Incorrect. If power is above P-9, the reactor is tripped. Plausible if setpoint for P-8, (loss of loop flow), of 35% is used, this would be the correct answer.
- C. Incorrect. Trip setpoint from the graph is 7.2. The action is to trip the reactor (above 50%) or turbine (below 50%). Plausible if graph is misapplied and power could be stabilized because condenser pressure is stabilizing.
- D. Incorrect. Trip setpoint from the graph is 7.2. The action is to trip the reactor (above 50%) or turbine (below 50%). Plausible if graph is misapplied and it thought a power reduction should be continued to recover condenser pressure.

Technical References: OP AP-7, attachment 2, OIM page B-6-2**References to be provided to applicants during exam:** AP-7, Attachment 2**Learning Objective:** 3477G Given an abnormal condition, summarize the major actions of OP AP-7 to mitigate an event in progress.

Question Source:	Bank #58 DCPN NRC 11/2012	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	

10CFR Part 55 Content:
Difficulty: 3.7

Comprehensive/Analysis
55.41.4

X

L162 Answer Key

Examination Outline Cross-Reference

APE 069 AK2.03 Knowledge of the interrelations between the Loss of Containment Integrity and the following: Personnel access hatch and emergency access hatch

Level	RO
Tier #	2
Group #	1
K/A #	APE 069 AK2.03
Rating	2.8

Question 61

Which of the following ensures Containment integrity is maintained while the unit is at power?

- A. Neither airlock door can be opened if the airlock seal pressure is low.
- B. After a time delay, an alarm is received in the Control Room if an airlock door is open for an extended period of time.
- C. A watch is stationed to immediately close the doors, if required, when both doors of an airlock are opened.
- D. Air leakage past the seals for the airlock doors is automatically tested each time the doors are operated and causes an alarm in the Control Room if leakage is unsatisfactory.

Proposed Answer: D. Air leakage past the seals for the airlock doors is automatically tested each time the doors are operated and causes an alarm in the Control Room if leakage is unsatisfactory

Explanation:

- A. Incorrect. Plausible there is an interlock but it prevents both doors from being open at the same time.
- B. Incorrect. Plausible, there is an alarm for a door open, but there is no time delay and a door open does not challenge containment integrity.
- C. Incorrect. Plausible, comp measures are sometimes allowed to take action if an automatic action is inoperable but this would not maintain containment integrity.
- D. Correct. An air test is done after a door is closed. If its excessive, PK11-12 alarms to alert the operator that leakage is potentially above what is allowed by technical specifications.

Technical References: AR PK11-12, AR PK11-24, LI-1

References to be provided to applicants during exam: None

Learning Objective: 9697I Apply TS 3.9 Technical Specification LCOs

Question Source: Bank #23 DCPD 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: 55.41.10
Difficulty: 3.7

Examination Outline Cross-Reference

EPE 074 EA2.03 Ability to determine or interpret the following as they apply to Inadequate Core Cooling: Availability of turbine bypass valves for cooldown

Level	RO
Tier #	1
Group #	2
K/A #	EPE 074 EA2.03
Rating	3.8

Question 62

GIVEN:

- The crew is performing the actions of EOP FR-C.1, Response to Inadequate Core Cooling
- RCPs have been secured by the operator
- All steam generators are intact and at approximately 1040 psig
- MSIVs are open
- All electrical buses are powered from startup
- RCS pressure is 1900 psig

The crew is preparing to depressurize the steam generators in order to inject accumulators.

In accordance with EOP FR-C.1, the operators will depressurize using the:

- A. 10% steam dumps at a rate not to exceed 100°F/hr.
- B. 10% steam dumps at a maximum rate without achieving a Main Steam Isolation.
- C. 40% steam dumps at a rate not to exceed 100°F/hr.
- D. 40% steam dumps at a maximum rate without achieving a Main Steam Isolation.

Proposed Answer: D. 40% steam dumps at a maximum rate without achieving a Main Steam Isolation.

Explanation:

Condenser is available – all buses are on startup and steam generators are intact, so the MSIVs will be open and the condenser steam dumps (40% steam dumps) will be used.

- A. Incorrect. Plausible – this would be the TS cooldown rate and not usually exceeded. In this instance, a more rapid cooldown is required to inject the accumulators and not allow conditions to degrade further.
- B. Incorrect. Plausible – this is the rate for using the 40% steam dumps.
- C. Incorrect. Plausible – the 40% steam dumps will be used, but this is the TS cooldown rate and not usually exceeded. In this instance, a more rapid cooldown is required to inject the accumulators and not allow conditions to degrade further.
- D. Correct. Using the 40% steam dumps is the preferred method and done at a rate to maintain the MSIVs open.

Technical References: EOP FR-C.1 step 12

References to be provided to applicants during exam: None

Learning Objective: 7920M Explain basis of emergency procedure steps (FR-Cs)

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

Past NRC Exam

Last Two NRC Exams

X

No

No

Question History:

Question Cognitive Level:

Memory/Fundamental
Comprehensive/Analysis

X

10CFR Part 55 Content:

55.41.5

Difficulty: 2.7

L162 Answer Key

Examination Outline Cross-Reference

E13 EA1.1 Ability to operate and / or monitor the following as they apply to the (Steam Generator Overpressure) Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Level	RO
Tier #	2
Group #	1
K/A #	E13 EA1.1
Rating	3.1

Question 63

GIVEN:

- The plant tripped from full power due to P-14, Steam Generator High-High Level Turbine Trip, actuation
- The affected steam generator narrow range level is now 100% and pressure is 1130 psig
- RCS temperature is 562°F
- Both reactor trip breakers are open
- The crew enters EOP FR-H.2, Response to Steam Generator Overpressurization

Step 2 of EOP FR-H.2 states:

2. VERIFY FW Isolation To Affected S/G:

- Mn Fdwtr Control Valves - CLOSED
- Mn Fdwtr Control Bypass Valves - CLOSED
- Mn Fdwtr Isol Vlvs - CLOSED

- 1) The operator should observe White Monitor light box lights OUT for _____.
- 2) For the current steam generator pressure the Steam Generator Safety Valve with the HIGHEST lift setpoint should be _____.

- 1) step 2c ONLY
2) closed
- 1) steps 2.a, 2.b and 2.c
2) closed
- 1) step 2.c ONLY
2) open
- 1) steps 2.a, 2.b and 2.c
2) open

Proposed Answer: D. 1) Steps 2.a, 2.b and 2.c 2) open

Explanation:

- Incorrect. P-14 closes all the valves listed. Because RCS temperature is above 554°F, its plausible to believe the Main Feedwater Reg and bypass are still open (closed by SI, P-14 or Low RCS Tave/P-4).
- Incorrect. All 5 safeties will be open above 1115 psig. Plausible – all the valves will be closed and the safety setpoints may not be known.
- Incorrect. All the valves will be closed. Plausible – all 5 safeties will be open.

D. Correct. All the valves are closed by P-14 and the highest setpoint for the safeties is 1115 psig, therefore, for the current pressure, all will be open.

Technical References: LC-2A, OIM B-6-12 , EOP FR-H.2

References to be provided to applicants during exam: None

Learning Objective: Analyze automatic features and interlocks associated with the Main Steam System. (7340)

Analyze automatic features and interlocks associated with the Main Feedwater System. (37614)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	
Difficulty: 3.3		

L162 Answer Key

Examination Outline Cross-Reference

E15 EK3.2 Knowledge of the reasons for the following responses as they apply to Containment Flooding: Normal, abnormal and emergency operating procedures associated with Containment Flooding.

Level	RO
Tier #	1
Group #	2
K/A #	E15 EK3.2
Rating	2.8

Question 64

Unit 1 has experienced a large break Loss of Coolant Accident.

The operator reports that containment recirc sump level is 99 feet. The STA reports that the Critical Safety Function Status tree for Containment Integrity is MAGENTA due to the sump level.

The crew will enter FR-Z.2, Response to Containment Flooding:

- A. to determine what additional water source is filling containment above design level.
- B. but immediately exit it once it is verified that this is approximately the expected level when the RWST reaches the transfer to cold leg recirc setpoint.
- C. to check for indications of a faulted steam generator.
- D. but immediately exit it once it is verified that this is approximately the expected level when the RWST reaches the level for containment spray termination.

Proposed Answer: A. to determine what additional water source is filling containment above design level.

Explanation:

- A. Correct. FRZ-2 step 1 directs operators to exit Flooding in Containment for recirc sump levels less than 95.75 feet. A level of 99 feet is much higher than what is expected for a LCOA. The Response to Containment Flooding Background Document, FRZ-2, states, "The maximum level of water in the containment following a major accident generally is based upon the entire water contents of the reactor coolant system, refueling water storage tank, condensate storage tank, and SI accumulators."
- B. Incorrect. The status tree may just be magenta during a LOCA but will not reach 99 feet. This is the water volume from the RCS, RWST, CST, and 4 SI Accumulators. This approximates the maximum water volume introduced following a LOCA plus a steamline or feedline break inside of containment.
- C. Incorrect. The correct level per step 1 of FRZ-2 is 95.75 feet. The maximum level of water in the containment following a major accident generally is based upon the entire water contents of the reactor coolant system, refueling water storage tank, condensate storage tank, and SI accumulators. This is the volume of LOCA plus a steamline or feedline break.
- D. Incorrect. The status tree may just be magenta during a LOCA but will not reach 99 feet. The correct level per step 1 of FRZ-2 is 95.75 feet.

Technical References: F-0, FR-Z.2 background, FR-Z.2

References to be provided to applicants during exam: None

Learning Objective:	Explain the basis of emergency procedure steps. (7920Q, R)	
Question Source:	Bank #63 DCPD NRC 11/2012	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	
Difficulty: 3.0		

L162 Answer Key

Examination Outline Cross-Reference

Level	RO
Tier #	1
Group #	2
K/A #	E08 EK1.3
Rating	3.5

E08 EK1.3 Knowledge of the operational implications of the following concepts as they apply to Pressurized Thermal Shock: Annunciators and conditions indicating signals, and remedial actions associated with Pressurized Thermal Shock.

Question 65

The crew is performing EOP FR-P.1, Response to Pressurized Thermal Shock.

- 1) In FR-P.1, an indication a soak is required is if RCS _____ leg temperatures exceed 100°F cooldown rate in any one hour.
 - 2) In accordance with FR-P.1, _____ would be allowed during the soak.
- A. 1) HOT
2) energizing pressurizer heaters to raise RCS pressure
 - B. 1) COLD
2) depressurizing to minimize RCS subcooling
 - C. 1) HOT
2) depressurizing to minimize RCS subcooling
 - D. 1) COLD
2) energizing pressurizer heaters to raise RCS pressure

Proposed Answer: B. 1) COLD 2) depressurizing to minimize RCS subcooling

Explanation:

- A. Incorrect. The cold legs are used because they are the best indication of downcomer temperature. During the soak, actions pressure is not allowed to increase during the soak.
- B. Correct. Cold legs are used because they are the best indication of downcomer temperature. The soak allows reduction of the thermal stress imposed by the cooldown on the reactor vessel. Allowable actions are those that do not cooldown the RCS or increase pressure.
- C. Incorrect. Cold legs are used, not the hot legs. The action would be allowed.
- D. Incorrect. Partially correct, cold legs are used, however, raising pressure is not allowed.

Technical References: FRR-P.1 step 24 and background

References to be provided to applicants during exam: None

Learning Objective: 4133 Explain RCS soak requirements

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5	

Difficulty: 2.7

Examination Outline Cross-Reference

Level	RO
Tier #	3
Group #	1
K/A #	G2.1.3
Rating	3.7

G2.1.3 - Knowledge of shift or short-term relief turnover practices.

Question 66

Both units are at 100% power.

- 1) In accordance with OP1.DC37, Plant Logs the _____ is responsible for completing the Shift Watch List.
- 2) If the minimum staffing requirements for Unit 2 cannot be met, the _____ is notified.
 - A. 1) Work Control SFM 2) Operations Manager
 - B. 1) Work Control SFM 2) Shift Manager
 - C. 1) Work Control Lead 2) Operations Manager
 - D. 1) Work Control Lead 2) Shift Manager

Proposed Answer: D. 1) Work Control Lead 2) Shift Manager

Explanation:

- A. Incorrect. While the WCSFM has an SRO license and maintains oversight, the WCL is *responsibility* of completing the Shift Watch list and while the Operations Manager has control of the department, the SM has the *responsibility* to ensure minimum staffing is met.
- B. Incorrect. The SM is notified but it's the WCL that fills out the Shift Watch List.
- C. Incorrect. The WCL is responsible, and while the Operations Manager has control of the department, the SM has the *responsibility* to ensure minimum staffing is met.
- D. Correct. The WCL is responsible to fill out the Shift Watch List and notify the SM is staffing is deficient.

Technical References: OP1.DC37

References to be provided to applicants during exam: None

Learning Objective: 7845 - Demonstrate the ability to apply administrative guide lines relative to shift staffing

Question Source:

(note changes; attach parent)

Bank #	
Modified Bank #	
New	X
Past NRC Exam	No
Last Two NRC Exams	No
Memory/Fundamental	X
Comprehensive/Analysis	
55.41.10	

Question History:

Question Cognitive Level:

10CFR Part 55 Content:

Difficulty: 2.0

Examination Outline Cross-Reference

G2.1.40 Knowledge of refueling administrative requirements.

Level	RO
Tier #	3
Group #	1
K/A #	G2.1.40
Rating	2.8

Question 67

In accordance with OP B-8DS2, Core Loading, which of the following is a responsibility of the RO in the Control Room during core loading?

- A. Coordinating Core Alterations
- B. Authorizing an unplanned deviation
- C. Maintaining the record of fuel movement
- D. Determining the cause of a Containment radiation monitor high alarm

Proposed Answer: C. Maintaining the record of fuel movement

Explanation:

- A. Incorrect. This is the responsibility of the Refueling SRO
- B. Incorrect. Responsibility of Reactor Engineering.
- C. Correct. Responsibility of RO in the Control Room
- D. Incorrect. Responsibility of Refueling SRO

Technical References: OP B-8DS2

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #67 DCPN NRC L091C 03/2012	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Difficulty: 2.0

Examination Outline Cross-Reference

Level	RO
Tier #	3
Group #	1
K/A #	G2.1.15
Rating	2.7

G2.1.15 - Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, Operations memos, etc.

Question 68

In accordance with OP1.DC31, Dissemination of Operations Information, which of the following types of communication would be used to inform crews of compensating measures for equipment operation for the next four weeks?

- A. Operations Awareness Bulletin
- B. Operations Standing Order
- C. Operations Shift Order
- D. Operations Policy

Proposed Answer: B. Operations Standing Order

Explanation:

- A. Incorrect. Awareness bulletins are addressed in OM15.ID11, Human Performance Event Response, used to share information about a plant event that is not a clock reset but is deemed important to share with the organization.
- B. Correct. Per OP1.DC31, Standing orders are longer term instructions (week or longer) and may include lessons learned, work arounds and compensatory measures
- C. Incorrect. Shift orders are short term instructions or communications for immediate use. Shift orders should not be utilized for dissemination of information which will exist long term (i.e., greater than one week). Longer term information should be included in the standing orders, or tracked by other appropriate means, such as procedure changes
- D. Incorrect. Policies are for all of operations and not signed off by an individual crew. There is no end date for a policy (has a revision number). Additionally, per OP1.DC31, crews are notified of a policy change by shift order.

Technical References: OP1.DC31, OM15.ID11

References to be provided to applicants during exam: None

Learning Objective: 46999 - Describe shift orders, standing orders, incident summaries, and operations policies, per OP1.DC31, section 4

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #67 DCPN NRC Exam 8/14	X
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

10CFR Part 55 Content:

55.41.10

Difficulty: 2.3

Examination Outline Cross-Reference

Level	RO
Tier #	3
Group #	2
K/A #	G2.2.43
Rating	3.0

G2.2.43 - Knowledge of the process used to track inoperable alarms.

Question 69

- 1) In accordance with OP1.DC37, Plant Logs, how often is the Control Operator required to review the open main annunciator problem evaluation sheets/defeat logs?
- 2) If a Control Room annunciator must be defeated at the Cable Spreading Room Maintenance Terminal, it is defeated by:
 - A. 1) Prior to assuming the watch 2) I & C
 - B. 1) Prior to assuming the watch 2) Operations
 - C. 1) Weekly 2) I & C
 - D. 1) Weekly 2) Operations

Proposed Answer: A. 1) Prior to assuming the watch 2) I & C

Explanation:

- A. Correct. The review is to be done at the beginning of each shift. If an alarm is defeated at the terminal in the CSR, it is performed by I& C.
- B. Incorrect. The review is to be done at the beginning of each shift. Operations does not defeat alarms at the terminal in the CSR. They can do alarms in the Control Room.
- C. Incorrect. An audit of the log is performed on a weekly basis, log is checked prior to assuming watch. Second part is correct.
- D. Incorrect. An audit of the log is performed on a weekly basis, log is checked prior to assuming watch. Operations does not defeat alarms at the terminal in the CSR. They can do alarms in the Control Room

Technical References: OP1.DC24 step 5.5.1.a., OP.DC37, MP I-1.5-9

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #70 DCPN NRC 6/2008	X
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Difficulty: 2.0

Examination Outline Cross-Reference

Level	RO
Tier #	3
Group #	2
K/A #	G2.2.13
Rating	4.1

G2.2.13 - Knowledge of tagging and clearance procedures.

Question 70

In accordance with OP2.ID2, Tagging Requirements, what type of tags are used by Operations to place equipment in a safe condition for maintenance, repair, or testing?

- A. Red tags
- B. Danger tags
- C. Caution tags
- D. Man-on-Line (MOL) tags

Proposed Answer: B. Danger tags

Explanation:

- A. Incorrect. Red tags are hung anytime a craft performs work associated within a clearance containing danger tags.
- B. Correct. Danger tags are used to protect personnel by tagging devices used to isolate sources of liquids, steam, gases, or electrical power or place equipment in a safe condition for maintenance, repair, or testing.
- C. Incorrect. Caution tags are not used to place equipment in a safe condition. They control operation and can be operated in accordance with the instructions on the tag.
- D. Incorrect. Like Danger tags, MOL tags are used to place equipment in a safe condition for maintenance, repair, or testing. However, MOL tags are only used for clearances involving non-plant equipment or by company utility personnel from outside Diablo Canyon Power Plant.

Technical References: OP2.ID2

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #71 DCPD NRC L121 08/2014	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	
Difficulty: 2.0		

Examination Outline Cross-Reference

Level	RO
Tier #	2
Group #	2
K/A #	G2.2.3
Rating	3.8

G2.2.3 - Knowledge of the design, procedural, and operational differences between units

Question 71

- 1) Unit 1 has _____ Control Bank A rods.
 - 2) Unit 2 has _____ Control Bank A rods.
- A. 1) Four 2) Four
- B. 1) Eight 2) Four
- C. 1) Four 2) Eight
- D. 1) Eight 2) Eight

Proposed Answer: B. 1) Eight 2) Four

Explanation:

- A. Incorrect. Unit 1 has eight CBA control rods and four CBB rods – the opposite of Unit 2 (4 CBA and eight CBB). This would be correct for Unit 1 CBB and Unit 2 CBA
- B. Correct. The Unit 1 CBA rods is correct, Unit 2 has four rods.
- C. Incorrect. Unit 1 CBA rods is eight (four for CBB), Unit 2 has eight CBB.
- D. Incorrect. CBA for Unit 1 has eight rods (four for Unit 2) and Unit 2 has eight CBB (4 on Unit 1).

Technical References: LA-3A

References to be provided to applicants during exam: None

Learning Objective: 5038 Describe unit differences associated with the Rod Control System

Question Source:	Bank #DCPP S-29786	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5	

Difficulty: 2.7

Examination Outline Cross-Reference

Level	RO
Tier #	3
Group #	3
K/A #	G2.3.15
Rating	2.9

G2.3.15 - Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Question 72

A Personal Electronic Dosimeter, (PED), for a watchstander in the Auxiliary Building will measure what type(s) of radiation?

- A. Gamma only
- B. Beta only
- C. Gamma and Beta only
- D. Neutron, Beta and Gamma

Proposed Answer: C. Gamma and Beta only

Explanation:

- A. Incorrect because a PED measures beta and gamma. Plausible because gamma is the most plentiful radiation present in a PWR aux building.
- B. Incorrect because a PED measures both beta and gamma. Plausible because beta is a common radiation present where leakage from radioactive systems occurs.
- C. Correct. A PED measures both beta and gamma, but not neutron radiation.
- D. Incorrect because a PED will not measure neutron radiation. Plausible because a TLD that can measure all 3 types can be issued, and it could be confused with the PED issued at the access point.

Technical References: Fundamentals - LFC7S

References to be provided to applicants during exam: None

Learning Objective: DESCRIBE the characteristics and principles of operation of each of the following types of portable/personal radiation monitoring instruments. (66027)

Personal electronic dosimeters (PED)

Question Source:	Bank #72 DCPN NRC 10-2016	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	Yes
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.11	

Difficulty: 2.0

Examination Outline Cross-Reference

Level	RO
Tier #	3
Group #	3
K/A #	G2.3.4
Rating	3.4

G2.3.4 - Knowledge of radiation exposure limits under normal or emergency conditions

Question 73

What is the Diablo Canyon TEDE Administrative Guideline for a male worker for a calendar year?

- A. 500 mrem
- B. 2000 mrem
- C. 4000 mrem
- D. 5000 mrem

Proposed Answer: B 2000 mrem

Explanation:

- A. Incorrect. Limit for declared pregnant worker is 10% of Federal limit 500 mrem
- B. Correct. 2000 mrem is the guideline
- C. Incorrect. The admin limit is 80% of the federal limit or 4000 mrem.
- D. Incorrect. 5000 mrem is the federal limit.

Technical References: RP1.ID6

References to be provided to applicants during exam: None

Learning Objective:

Question Source: (note changes; attach parent)	Bank #71 DCPN NRC 06/2008 Modified Bank # New Past NRC Exam	X
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content: Difficulty: 2.0	55.41.12	

Examination Outline Cross-Reference

Level	RO
Tier #	3
Group #	3
K/A #	G2.4.25
Rating	3.3

G2.4.25 -Knowledge of fire protection procedures.

Question 74

A fire in the Auxiliary Building has been reported to the Control Room.

The fire brigade has been dispatched to the scene of the fire.

Which two of the following functions does the designated OPS Responder perform per CP M-6, Site Fire Response?

1. Takes a direct role in the fire fighting efforts.
2. Performs emergency notifications to offsite personnel.
3. Communicates with the control room on status of fire fighting efforts.
4. Notifies Incident Commander and Radiation Protection of changes in ventilation.

- A. 1 and 2
- B. 1 and 4
- C. 2 and 3
- D. 3 and 4

Proposed Answer: D. 3 and 4

Explanation:

- A. Incorrect. All actions are taken for a fire, however, the OR does not to enter the fire area or make notifications offsite (keeps CR informed).
- B. Incorrect. OR not to enter the fire area, but notifies the IC and RP of ventilation changes.
- C. Incorrect. Does not perform offsite notifications, but does communicate with the CR.
- D. Correct. Both are duties of the OR (step 3 and 7 of attachment 7.2)

Technical References: CP M-6, attachment 7.2

References to be provided to applicants during exam: None

Learning Objective:

Question Source: (note changes; attach parent)	Bank #74 DCPN NRC Exam 06/2008 Modified Bank # New Past NRC Exam	X Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content: Difficulty: 2.0	55.41.10	

Examination Outline Cross-Reference

G2.4.19 -Knowledge of EOP layout, symbols, and icons.

Level	RO
Tier #	3
Group #	4
K/A #	G2.4.19
Rating	3.4

Question 75

When proceeding through Emergency Operating Procedures, a step number preceded by a diamond (◊) , denotes that the step:

- A. impacts a Time Critical Operator Action
- B. appears on the procedure foldout page
- C. is an Immediate Action
- D. is a Continuous action

Proposed Answer: A. impacts a Time Critical Operator Action

Explanation:

- A. Correct. A TCOA step has a diamond by the step number and denotes it impacts the TCOA completion time
- B. Incorrect. Foldout page items are not identified in steps of the procedure
- C. Incorrect. A box around the step number indicates the step is an immediate action.
- D. Incorrect. A continuous action is denoted by enclosing the entire step within a box

Technical References: E-0, AD1.DC12

References to be provided to applicants during exam: None

Learning Objective: Describe the expectations and standards for abnormal procedure use and adherence, including: (41678)

- Performance of immediate actions

Question Source:	Bank #73 DCPN NRC Exam 03/2012	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Difficulty: 2.0

Examination Outline Cross-Reference

EPE 009 G2.1.7 Small Break LOCA: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Level	SRO
Tier #	1
Group #	1
K/A #	EPE 009 G2.1.7
Rating	4.7

Question 76

The crew has initiated Safety Injection based on the following:

- RCS leak estimated at 150 gpm
- RM-11, Containment Air Particulate and RM-12, Containment Rad Gas, are in high alarm
- PK11-21, High Radiation, in alarm

While performing E-0, Reactor Trip or Safety Injection, the operator reports the following:

- RM-74, Steamline Radiation Monitor, has just pegged high, and PK11-18, Main Steam Line Hi Rad, has alarmed
- RM-15 and RM-15R, Steam Jet Air Ejector Radiation Monitors, have both lowered from their normal, pre-trip levels
- RM-19/23, Steam Generator Blowdown Radiation Monitors, have remained at pre-trip levels
- All steam generator narrow range levels are approximately 40% and rising slowly

Which of the following procedure flowpaths from E-0, should be taken by the Shift Foreman to mitigate the event?

- Go to E-1, Loss of Reactor or Secondary Coolant, and remain in E-1.
- Go to E-1, Loss of Reactor or Secondary Coolant, then transition to E-1.2, Post LOCA Cooldown and Depressurization.
- Go to E-3, Steam Generator Tube Rupture, then transition to E-3.1, Post SGTR Cooldown Using Backfill.
- Go to E-3, Steam Generator Tube Rupture, then transition to ECA-3.1, SGTR With Loss of Reactor Coolant-Subcooled Recovery Desired.

Proposed Answer: B. Go to E-1, Loss of Reactor or Secondary Coolant, then transition to E-1.2, Post LOCA Cooldown and Depressurization. QUESTION DELETED. SEE EXAM REPORT 2018-301 FOR DETAILS

Explanation:

- Incorrect. The leak size is such that E-1.2 will be used to cooldown the RCS and perform the step wise ECCS flow reduction. E-1 is for LOCAs that reduce RCS pressure to less than RHR injection pressure, 300 psig.
- Correct. Diverse indications of SG activity should convince the operators that RM-74 has failed high. There is no valid indication of activity in any SG. E-0 diagnostic should lead to E-1. With a leak rate of only 150 gpm, the next procedure transition at Step 13 should be E-1.2.
- Incorrect. There is not a valid indication of a tube rupture. E-0 transition is "valid" alarm. Plausible, this would be correct if there was a tube rupture and RCS leakage.
- Incorrect. No valid indication of a tube rupture. Plausible, this is the normal flowpath for a tube rupture.

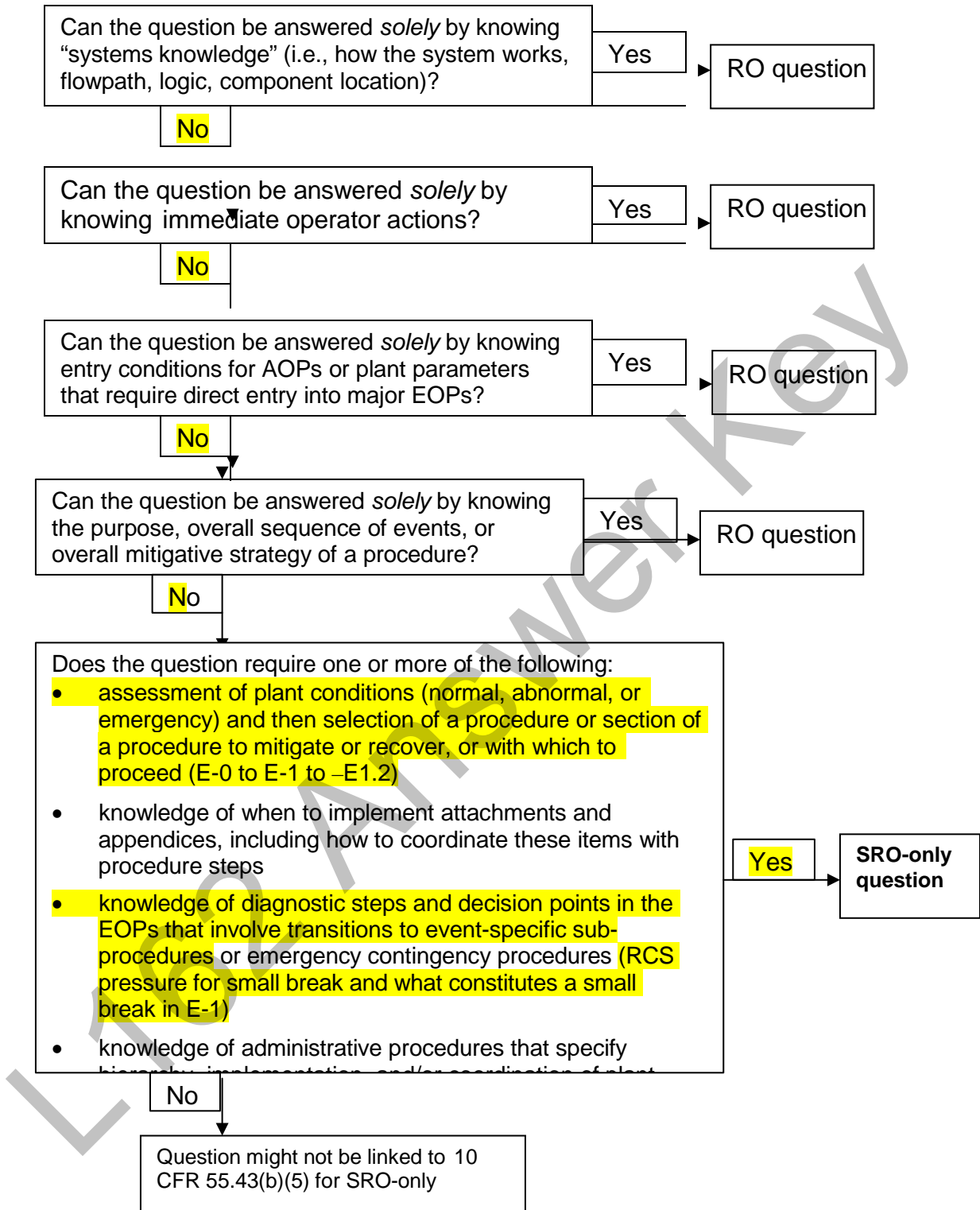
Technical References: E-0, E-1, E-1 background

References to be provided to applicants during exam: none

Learning Objective:		
Question Source: (note changes; attach parent)	Bank #DCPP Bank P-91788 Modified Bank # New	X
Question History:	Past NRC Exam Last Two NRC Exams	No No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 X
10CFR Part 55 Content: Difficulty: 3.0	55.43.5	

L162 Answer Key

SRO Justification:



Examination Outline Cross-Reference

APE 022 G2.4.4 – Loss of Reactor Coolant Makeup: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Level	SRO
Tier #	1
Group #	1
K/A #	APE022 G2.4.4
Rating	4.7

Question 77

RCS temperature is 400°F and pressure is 600 psig.

The operator reports the following:

- Pressurizer level is lowering
- Pressurizer backup heaters are OFF
- Charging flow indicates 0 gpm
- Seal injection is 0 gpm to each RCP
- CCP 1-3 amps are approximately 60 amps
- FCV-128, CCP Flow Control Valve, demand indicates 100%
- Letdown is in service at 75 gpm
- VCT level is lowering

The Shift Foreman will go to OP AP-17, Loss of Charging:

- A. Section A, Loss of All Charging.
- B. Section B, Charging System Equipment Malfunctions.
- C. Section C, Charging Line Leak at Power.
- D. Section D, Charging Line Leak on RHR.

Proposed Answer: C. Section C, Charging Line Leak at Power.

Explanation:

Event is a loss of charging due to a leak. Indications are for a leak - lowering VCT and pressurizer level indicates a leak. Multiple indications for each section, some overlapping, cause the SRO to determine which section will successfully deal with the event in progress.

- A. Incorrect. While charging flow indication is zero, it is not a loss of charging but a leak which is another procedure section.
- B. Incorrect. While FCV-128 indicates full open and flow is 0 gpm, it is not a malfunction of a component, but a leak that has occurred as indicated by lowering VCT and pressurizer level.
- C. Correct. Pressurizer and VCT level are lowering due to leak upstream of FCV-128.
- D. Incorrect. The plant is in MODE 3, RHR would not in service, but there is a leak.

Technical References: OP AP-17

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 #82 DCPN NRC Exam 03/2012	X
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No

Question Cognitive Level:

Memory/Fundamental
Comprehensive/Analysis

X

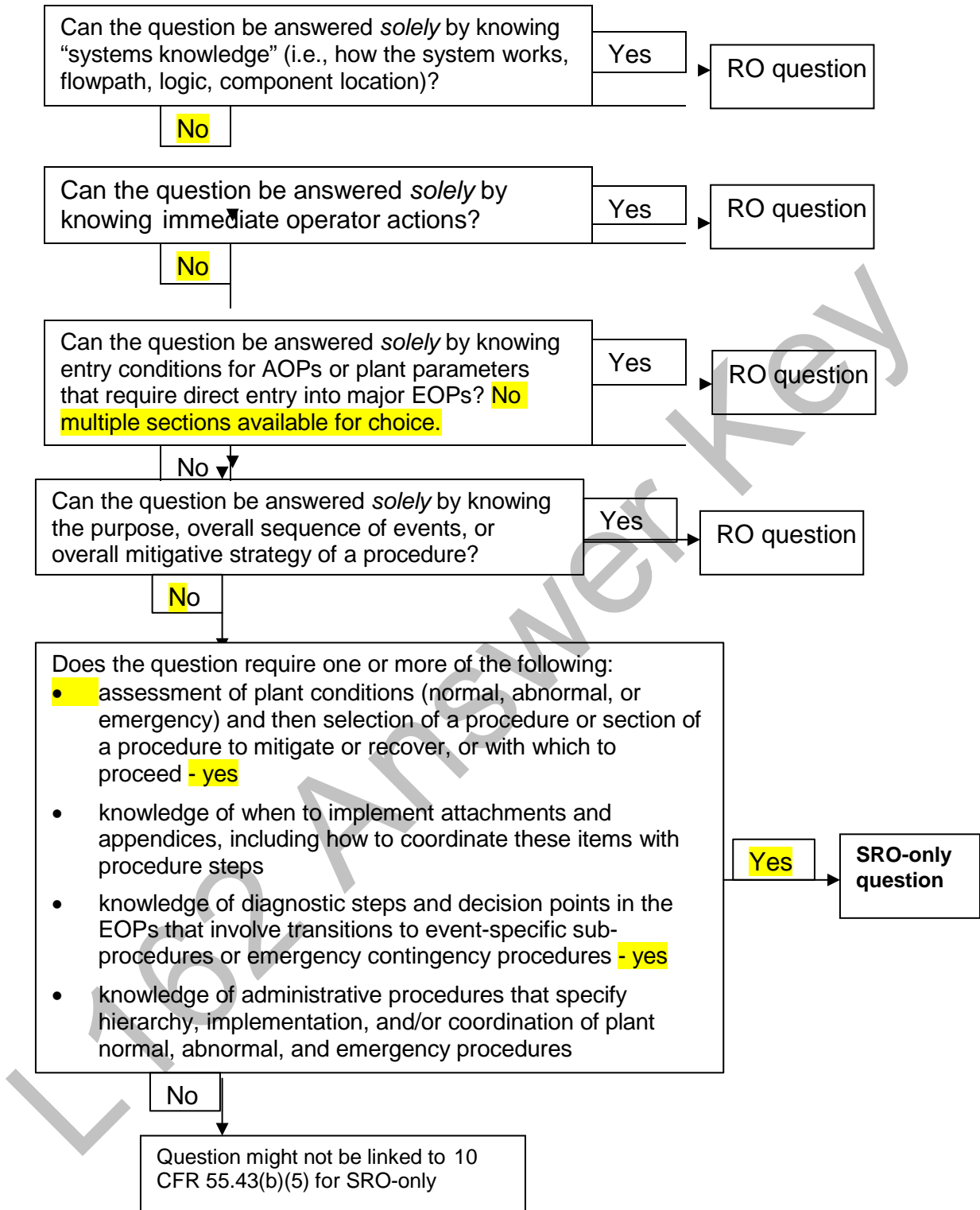
10CFR Part 55 Content:

55.43.5

Difficulty: 3.0

L162 Answer Key

SRO Justification:



Examination Outline Cross-Reference

EPE 029 EA2.08 Ability to determine or interpret the following as they apply to a ATWS: Rod bank step counters and RPI.

Level	SRO
Tier #	1
Group #	1
K/A #	EPE 029 EA2.08
Rating	3.5

Question 78

GIVEN:

- Unit 1 experienced an RCS leak
- Safety Injection actuated
- In E-0, Reactor Trip or Safety Injection, the reactor failed to automatically or manually trip
- The crew is performing EOP FR-S.1, Response To Nuclear Power Generation – ATWS
- In accordance with FR-S.1, the Work Control Lead has implemented and completed E-0, steps 1 through 4 and steps 1 through 11 of Appendix E

Later, while performing step #9 of FR-S.1, “CHECK If Reactor Is Subcritical”, the operator reports:

- Reactor Trip Breakers are closed
- All DRPI rod bottom lights are LIT
- Group Step counters for Control Bank read 110 steps
- Intermediate Range Start Up rate is negative

What action should be taken by the Shift Foreman?

- Stop at the current step of FR-S.1 and go to E-0, step 1.
- Stop at the current step of FR-S.1 and go to E-0, step 5.
- Prepare to exit the procedure by going to step 19, Ensure Adequate Shutdown Margin, then go to E-0, step 1.
- Prepare to exit the procedure by going to step 19, Ensure Adequate Shutdown Margin, then go to E-0, step 5.

Proposed Answer: C. Prepare to exit the procedure by going to step 19, Ensure Adequate Shutdown Margin, then go to E-0, step 1.

Explanation:

- Incorrect. The reactor is subcritical, however, the procedure is exited at a defined point and then the proper transition is to E-0, step 1 despite having had the steps done while in S.1.
- Incorrect. The reactor is subcritical, however, the procedure is exited at a defined point and then the proper transition is to E-0, step 1 despite having had the steps done while in S.1.
- Correct. With the reactor subcritical, the action will be to complete FR-S.1 to a defined exit point and then go to E-0, step 1.
- Incorrect. While some E-0 steps have been implemented and completed, E-0 still must be addressed and the diagnostic steps performed to ensure the appropriate EOP is transitioned to from E-0.

Technical References: FR-S.1

References to be provided to applicants during exam: none

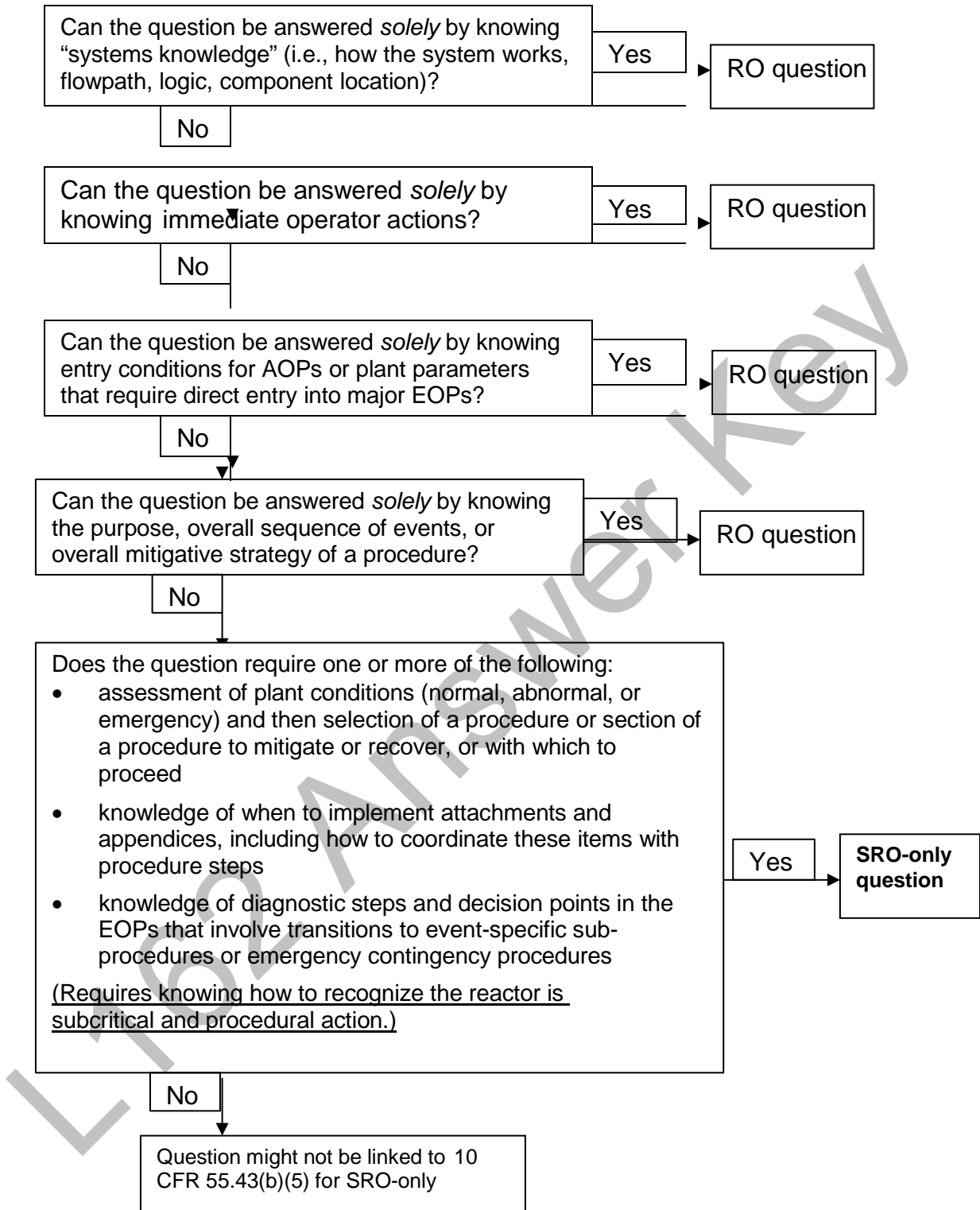
Learning Objective: 9703 Identify exit conditions for the FRPs

Question Source: Bank #81 South Texas Project 04/2014 X
(note changes; attach parent) Modified Bank #

Question History:	New	
	Past NRC Exam	Yes
Question Cognitive Level:	Last Two NRC Exams	No
	Memory/Fundamental	
10CFR Part 55 Content:	Comprehensive/Analysis	X
Difficulty: 2.5	55.43.5	

L162 Answer Key

SRO Justification:



Examination Outline Cross-Reference

Level	SRO
Tier #	1
Group #	1
K/A #	APE 054 G2.1.23
Rating	4.4

APE 054 G2.1.23 Loss of Main Feedwater: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Question 79

The crew is stabilizing the plant at approximately 50% power following a program ramp from 90% power in accordance with OP AP-15, Loss of Feedwater Flow, due to the trip of a Main Feed pump.

In accordance with OP AP-15, Chemistry is notified of the power change because a sample is required in the next 4 to 6 hours to satisfy:

- A. SR 3.4.16.1, Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 600.0 $\mu\text{Ci/gm}$.
- B. SR 3.4.16.2, Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 $\mu\text{Ci/gm}$.
- C. SR 3.7.18.1, Verify the specific activity of the secondary coolant is ≤ 0.10 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131.
- D. SR 7.4 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters in Table 7.4-1.

Proposed Answer: B. SR 3.4.16.2, Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 $\mu\text{Ci/gm}$.

Explanation:

- A. Incorrect. Required per SR 3.4.16.1 but not after a power change.
- B. Correct. Required per SR 3.4.16.2 - Between 2 and 6 hours after a THERMAL POWER change of $\geq 15\%$ RTP within a 1 hour period.
- C. Incorrect. steam generator specific activity is addressed by SR 3.7.18.1 but not required after a power change.
- D. Incorrect. Addressed by ECG 7.4, SR 7.4 but not required for power changes (every 72 hours).

Technical References: OP AP-15, LCO 3.4.16, 3.7.18, ECG 7.4

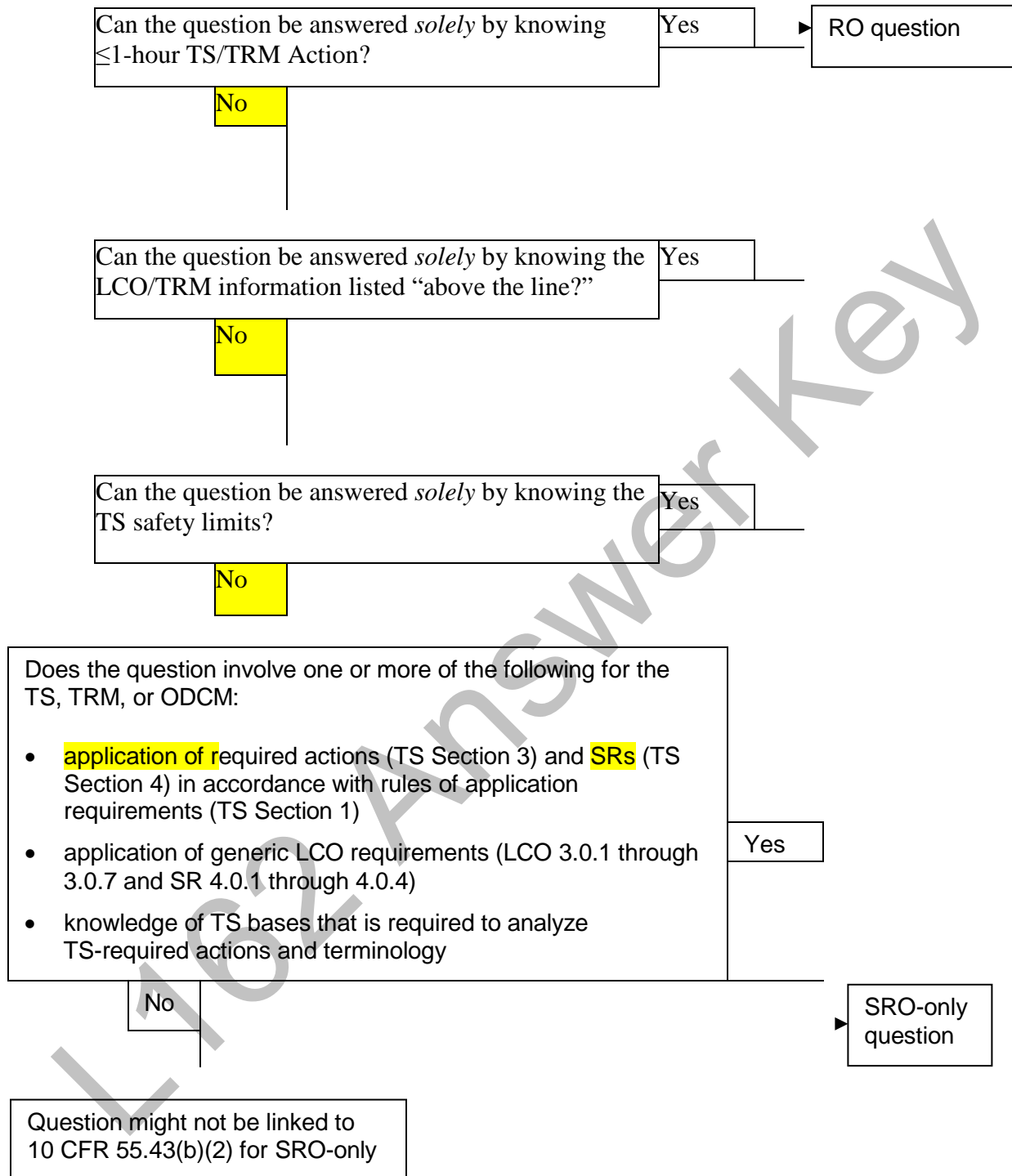
References to be provided to applicants during exam: none

Learning Objective: 9697D - Apply TS 3.4 Technical Specification LCOs

Question Source:	Bank #79 DCPN NRC L051 04/2007	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.2	

Difficulty: 2.5

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Examination Outline Cross-Reference

APE 058 AA2.03 Ability to determine and interpret the following as they apply to the Loss of DC Power: DC loads lost; impact on ability to operate and monitor plant systems.

Level	SRO
Tier #	1
Group #	1
K/A #	APE 058 AA2.03
Rating	3.9

Question 80

The crews for both units have entered ECA-0.0, Loss of All Vital AC Power.

If the loss of power is determined to last for a MINIMUM of 1), then the crews should 2) FSG 04, DC Bus Load Shed and Management to extend the availability of the vital batteries.

- A. 1) one hour 2) REFER TO
- B. 1) one hour 2) IMPLEMENT
- C. 1) two hours 2) REFER TO
- D. 1) two hours 2) IMPLEMENT

Proposed Answer: B. 1) one hour 2) IMPLEMENT

Explanation:

NOTE:

REFER TO: The designated procedure may be used for guidance at the Operator's discretion.

IMPLEMENT: Perform the second procedure in parallel with the procedure in effect Use of the FSG is SRO knowledge, per the FSG "With careful evaluation, directed by the SM/SEC, this guideline can be modified to suit the plant conditions existing at the time of its use. Additionally, whether the procedure is performed or implemented is not required RO knowledge (that there is a procedure to address DC load shed, would be RO knowledge, not the type of adherence required). The knowledge of how quickly to implement FSG 04 will aid the operating crew in the ability to operate and monitor the minimum key systems and indications associated with DC on a loss of all power event..

- A. Incorrect. An ELAP is a loss of power for greater than 1 hour. The FSGs are implemented from ECA-0.0.
- B. Correct. The FSG is IMPLEMENTED if power is lost for greater than 1 hour.
- C. Incorrect. The procedure is implemented after 1 hour. 2 hours is plausible as this is the design basis for the batteries.
- D. Incorrect. The procedure is implemented, 2 hours is plausible as this is the design basis for the batteries.

Technical References: ECA-0.0, FSG 04, OP1.DC10

References to be provided to applicants during exam: none

Learning Objective: Explain the basis for securing DC loads on loss of vital AC bus. (7118)

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

Past NRC Exam

Last Two NRC Exams

X

No

No

Question History:

Question Cognitive Level:

Memory/Fundamental
Comprehensive/Analysis

X

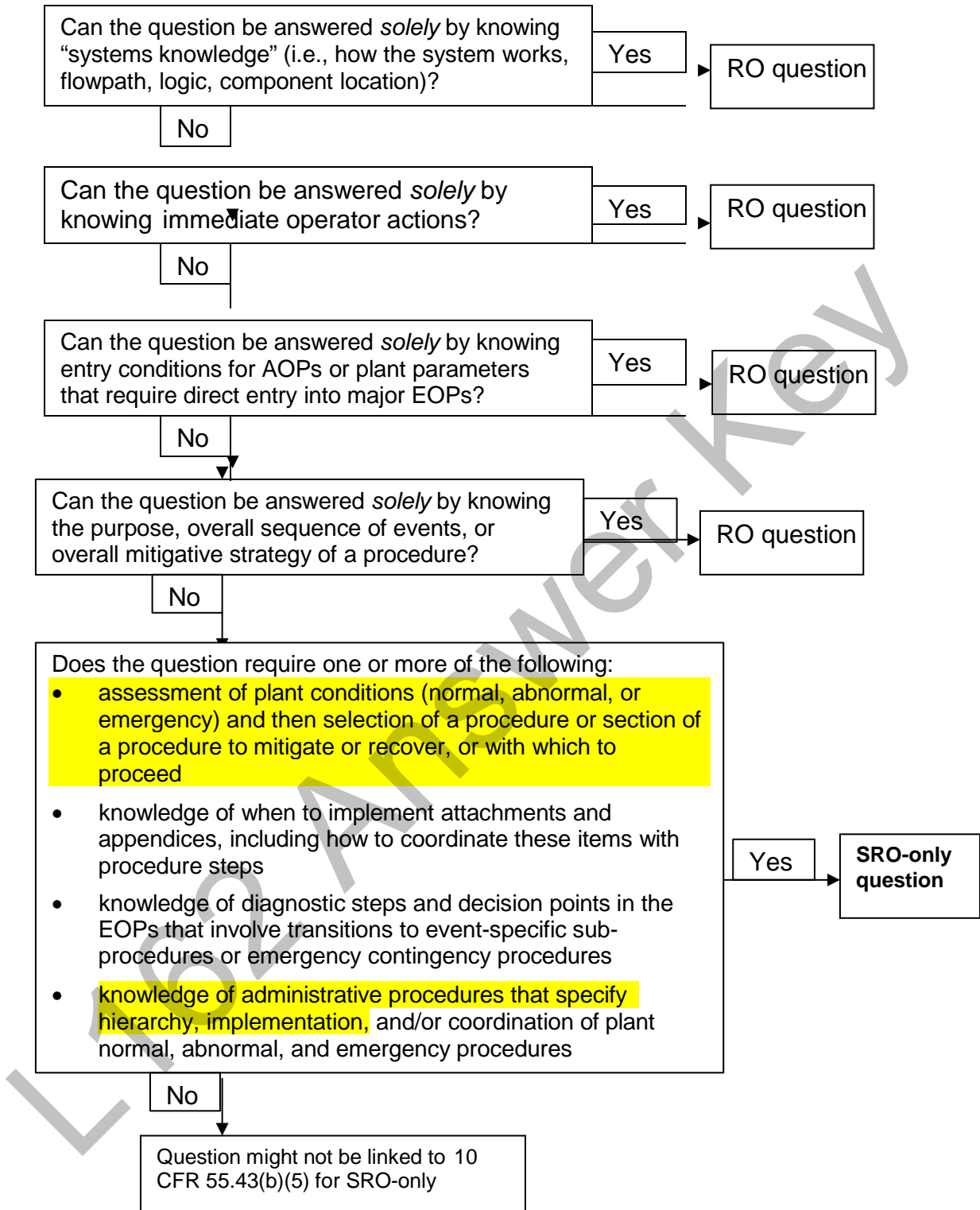
10CFR Part 55 Content:

55.43.5

Difficulty: 2.0

L162 Answer Key

SRO Justification:



Examination Outline Cross-Reference

APE 077 AA2.05 Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: Operational status of offsite circuit.

Level	SRO
Tier #	1
Group #	1
K/A #	APE 077 AA2.05
Rating	3.8

Question 81

GIVEN:

- Both units are at 100% power
- An earthquake causes a grid disturbance
- Grid Control reports:
 - loss of the Morro Bay-Diablo Canyon 230 kV line
 - loss of the Diablo Canyon-Gates 500 kV line
 - 500 kV voltage is unchanged
 - 230 kV voltage is 232 kV
 - Los Padres Area Load is 420 MW
 - DCPD has 2 capacitors IN
 - Mesa transformer is in service
 - Mesa has 2 capacitors that are available and IN
- Auto start of all standby Condensate/Booster Pump Sets is blocked

What is the operational status of the 230 kV and 500 kV offsite power sources?

- A. Both the 230 kV and 500 kV offsite power sources remain OPERABLE.
- B. The 230 kV source remains OPERABLE, the 500 kV source is inoperable.
- C. The 230 kV source is inoperable, the 500 kV source remains OPERABLE.
- D. Both the 230 kV and 500 kV offsite power sources are inoperable.

Proposed Answer: C. The 230 kV source is inoperable, the 500 kV source remains OPERABLE.

Explanation:

The SFM is responsible for Tech Spec entries due to inoperable electrical sources or equipment. If there is a question of operability, SR 3.8.1.1 is performed by completing section 17 of STP I-1C and OP J-2:VIII.

Working thru the attachments of OP J-2:VIII:

For DCPD/Mesa configuration, table 1, 2 DCPD caps - 2 and 2 caps for Mesa is line 2-3.

Minimum DCPD normal transfer limits - voltage 230 kV , load 400 MW. For Level C, 229 and 415 respectively

Per attachment 2 (left to right) – YES for first block (230 required but currently 232), NO for next block (Normal transfer load limit is 400 and currently at 420 MW). Configuration (at power) 3 for next block, Comp actions 1 or 3 complete – YES (Cond Booster pumps auto start blocked), YES for next block (Level C voltage greater than 229), Load less than Level C – NO, load is 420, max is 415 mw. Startup is inoperable.

- A. Incorrect. There are two 230 kV lines and three 500 kV lines. According to the bases for LCO 3.8.1, the sources are OPERABLE if one 230 kV (with conditions) and two 500 kV lines satisfy the LCO. A loss of one line of each does not affect operability providing

voltage and load for the existing 230 kV lineup is acceptable. For the current lineup, the minimum voltage is 229 kV. Working thru the attachments, show that this is the minimum even if comp measures are in place. NOTE: The 12 kV cutouts would need to be taken to cutout, the condensate booster pump cutouts are cutout at power. For the current conditions, voltage is too low, the 230 kV is inoperable. 500 kV voltage has not changed and because a loss of a single line does not make the system inoperable, the 500 kV system remains OPERABLE.

- B. Incorrect. A single line loss does not make the 500 kV inoperable as long as voltage is acceptable, if its unchanged from the initial conditions, then the 500 kV is still OPERABLE. Plausible because loss of one line would seem likely to cause at least one offsite source inoperable. Additionally, using J-2:VIII, 230 kV is inoperable.
- C. Correct. A single line loss does not make 500 kV inoperable. Additionally, for the current configuration, 230 kV is inoperable. See explanation above.
- D. Incorrect. A single line loss does not make 500 kV inoperable. Plausible because 230 kV is inoperable and loss of a line would seem plausible to be inoperable.

Technical References: LCO 3.8.1, OIM J-1-1 and OP J-2:VIII

References to be provided to applicants during exam: OP J-2:VIII att 1, 2 and 49 and 66 (page 37 and 54 of 64 of attachment 3)

Learning Objective: 9697H Apply TS 3.8 Technical Specification LCOs

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Past NRC Exam

No

Question History:

Last Two NRC Exams

No

Question Cognitive Level:

Memory/Fundamental

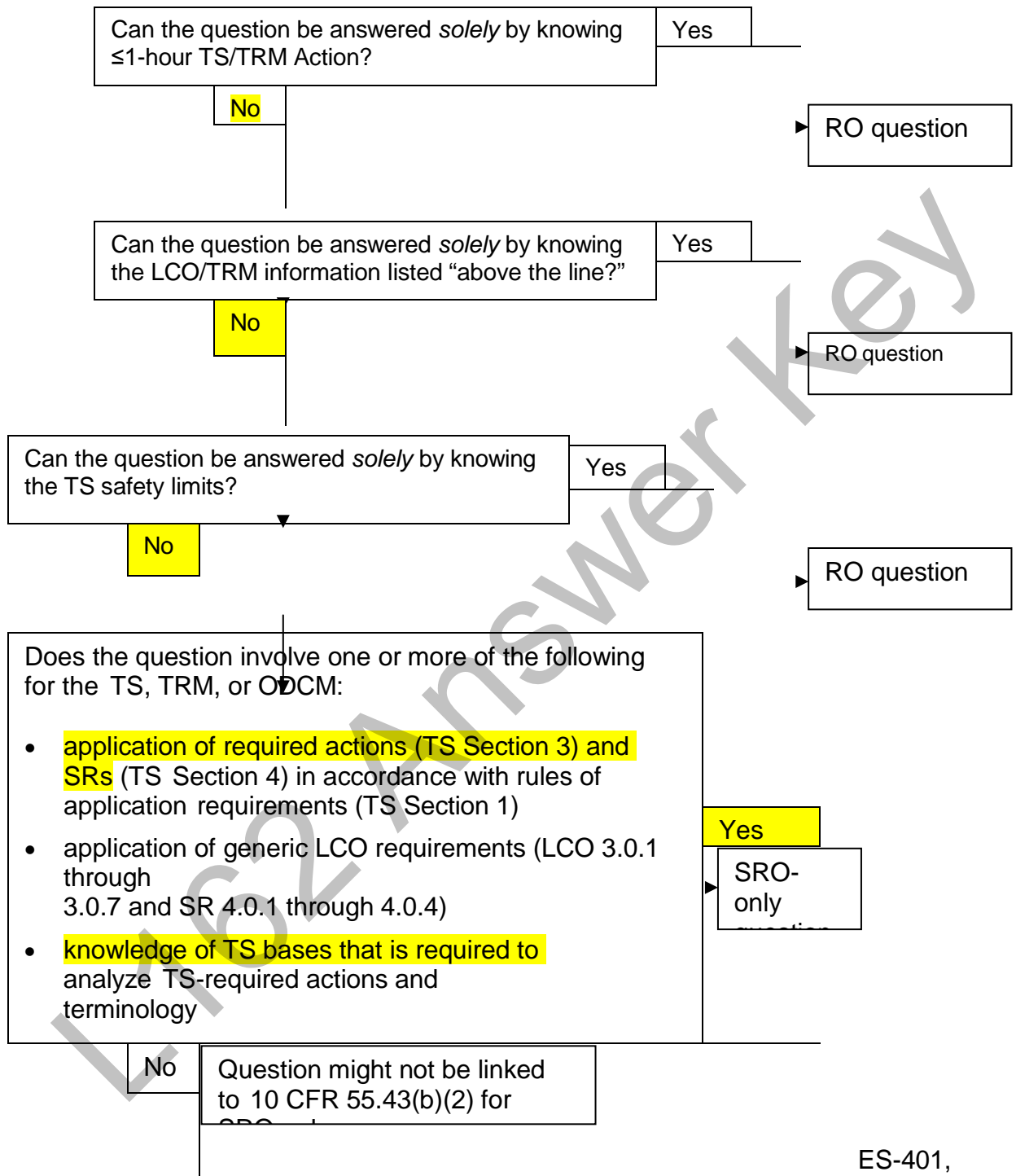
Comprehensive/Analysis

X

10CFR Part 55 Content:

55.43.2

Difficulty: 3.5



Examination Outline Cross-Reference

APE033 AA2.04 Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Satisfactory overlap between source-range, intermediate-range and power-range instrumentation.

Level	SRO
Tier #	1
Group #	2
K/A #	APE 033 AA2.04
Rating	3.2

Question 82

A plant shutdown is in progress. Reactor power is 25%.

A loss of compensating voltage to Intermediate Range channel N36 occurs.

- 1) During the shutdown, P6, Intermediate Range Permissive _____ automatically energize the Source Range channels
 - 2) The Shift Foreman is required to enter a Technical Specification LCO for the inoperable N36 channel _____.
- A. 1) will
2) immediately
- B. 1) will
2) when reactor power is less than 10%
- C. 1) will NOT
2) immediately
- D. 1) will NOT
2) when reactor power is less than 10%

Proposed Answer: D.

- 1) will NOT
- 2) when reactor power is less than 10%

Explanation:

Candidate must know that due to the loss of compensating voltage, the channel will not lower sufficiently to provide the necessary overlap and cause P6 to automatically energize the Source Range (requires two IR channels less than 10^{-10} amps. Also, from LCO 3.3.1 that the action is to either lower power to less than P-6 or raise power to greater than 10% with one channel inoperable. Coming down, the LCO will not be entered until power is less than 10% (SRO).

- A. Incorrect. For many instrument failures, the LCO immediately applies, such as for a failed Power Range channel. Also, the necessary overlap to energize the Source Range would be met if the N36 channel was over compensated (it would read lower and get to the setpoint sooner).
- B. Incorrect. The effect of undercompensation is to make the channel read higher, preventing auto energizing of the source range channels. Second part is correct..
- C. Incorrect. First part is correct, Second part plausible, for many instrument failures, a shutdown is required, however the LCO for IR channels apply when less than 10%
- D. Correct. P6 will not energize the source range channels and the LCO does not apply until less than 10% power.

Technical References: LCO 3.3.1 ACTION F, OIM B-4-3a

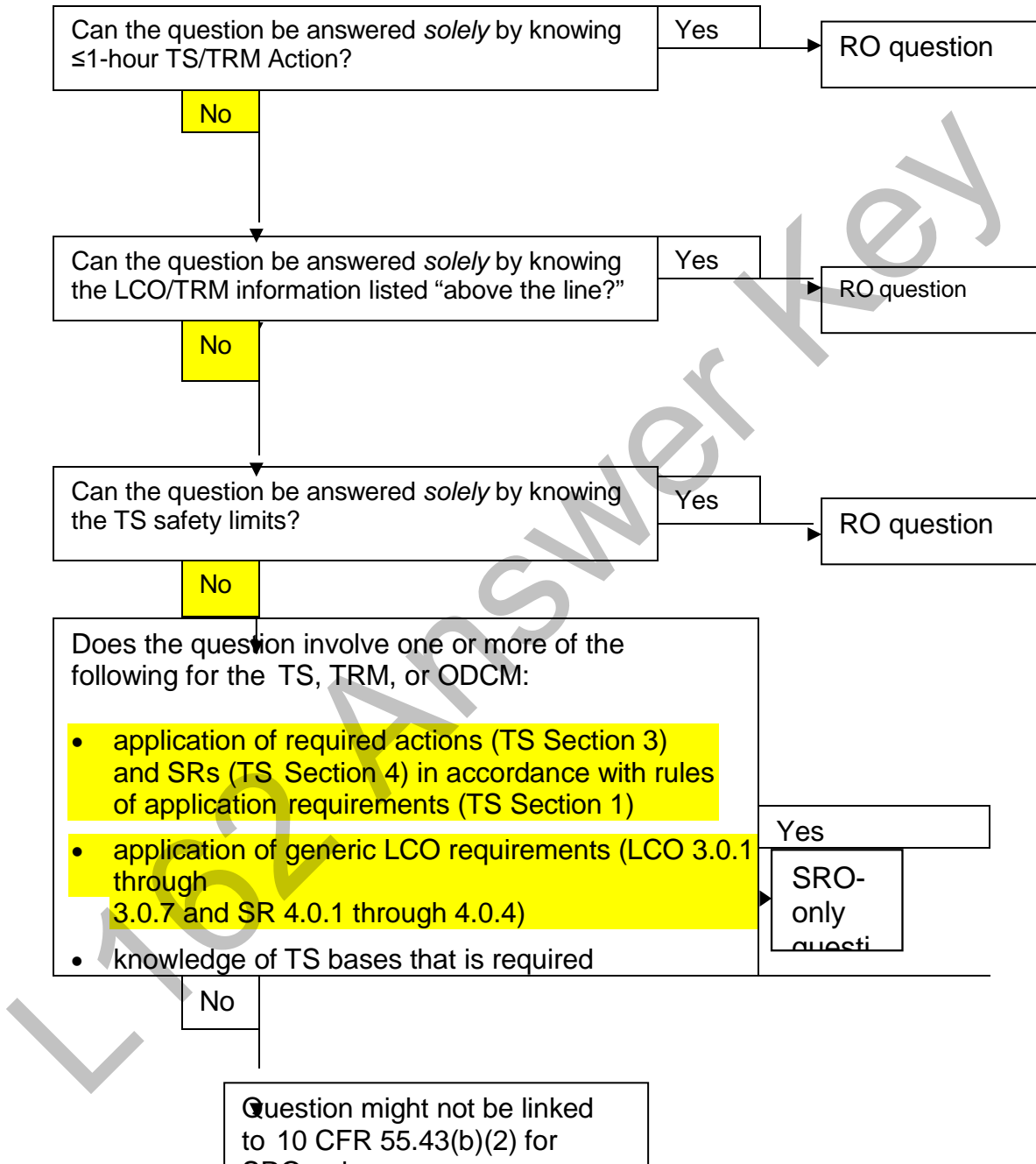
References to be provided to applicants during exam: none

Learning Objective:

Question Source: (note changes; attach parent)	Bank # Modified Bank # New Past NRC Exam Last Two NRC Exams	X No No
Question History:		
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.2	
Difficulty: 3.0		

L162 Answer Key

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Examination Outline Cross-Reference

APE 036 AA2.03 Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: Magnitude of potential radioactive release.

Level	SRO
Tier #	1
Group #	2
K/A #	APE036 AA2.03
Rating	4.2

Question 83

Per the bases for LCO 3.9.7, Refueling Cavity Water Level, offsite doses for a fuel handling accident inside containment are maintained well within allowable limits due to which of the following?

A minimum water level of 23 feet above the:

- A. top of a fuel assembly being moved and one loop of RHR in service to maintain RCS temperature less than 200°F.
- B. top of a fuel assembly being moved and a minimum decay time of 100 hours prior to fuel handling.
- C. reactor vessel flange and one loop of RHR in service to maintain RCS temperature less than 200°F.
- D. reactor vessel flange and a minimum decay time of 100 hours prior to fuel handling.

Proposed Answer: D. reactor vessel flange and a minimum decay time of 100 hours prior to fuel handling.

Explanation:

- A. Incorrect. Plausible - There is a requirement for the SFP for water level that maintains at least 8 feet above the top of the fuel for shielding (not fuel accident). The bases for the RHR in service LCO states the reason is that if the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the RHR System is required to be operational in MODE 6, with the water level \geq 23 ft above the top of the reactor vessel flange, to prevent this challenge.
- B. Incorrect. The requirement is water over the reactor vessel flange, not the top of the fuel, the decay time of 100 hours is correct.
- C. Incorrect. The level is correct. The RHR in service is to prevent boiling (which could reduce level).
- D. Correct. Per the bases for LCO 3.9.7, with a minimum water level of 23 ft and a minimum decay time of 100 hours 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.

Technical References: LCO Bases for 3.9.5, 3.9.7 and 3.7.15

References to be provided to applicants during exam: none

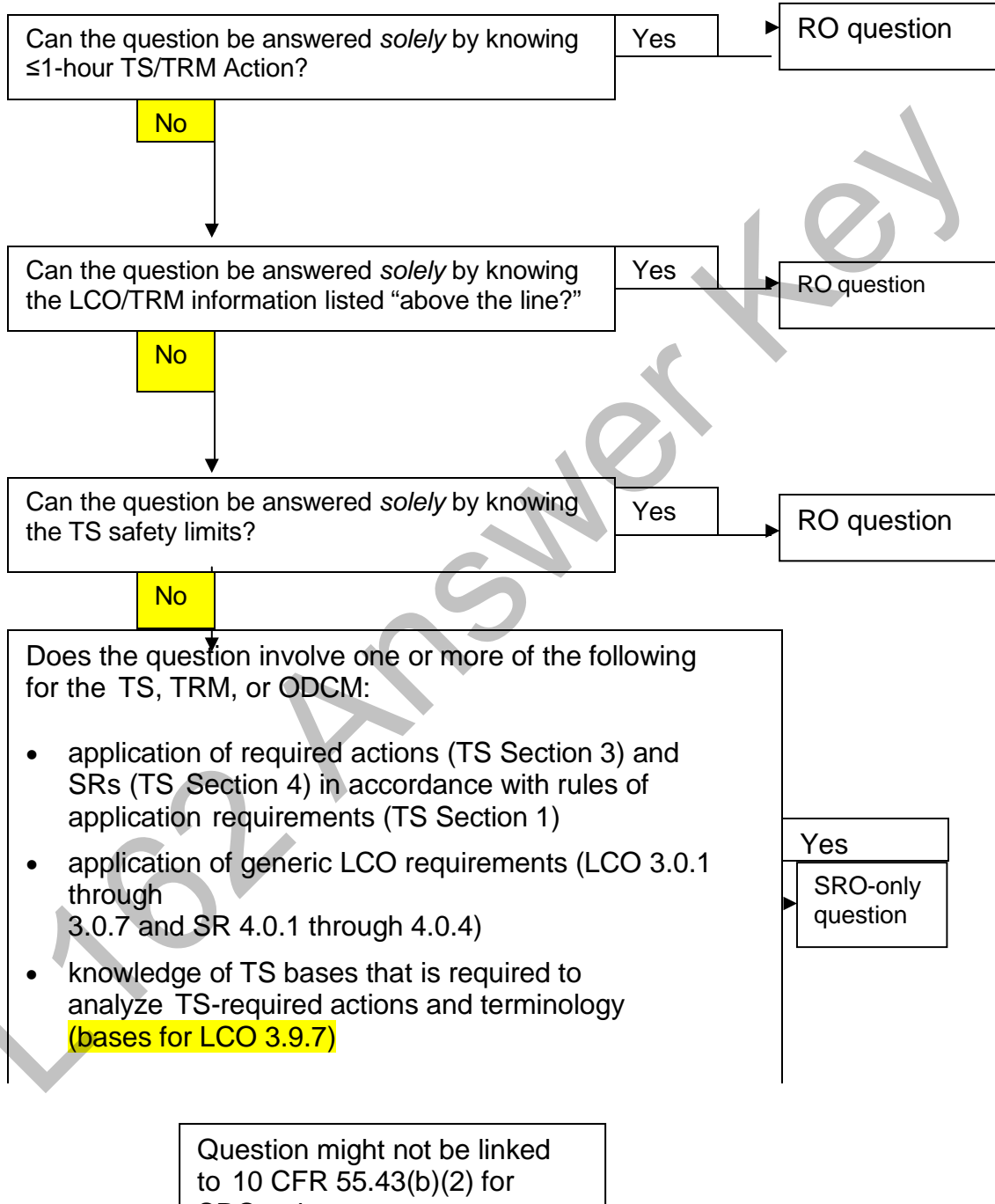
Learning Objective: 9694I - Apply TS 3.9 Technical Specification bases

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:	55.43.2	
Difficulty: 3.0		

L162 Answer Key

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Examination Outline Cross-Reference

**APE 037 G2.2.22 Steam Generator Tube Leakage:
Knowledge of limiting conditions for operations and safety
limits.**

Level	SRO
Tier #	1
Group #	2
K/A #	APE037 G2.2.22
Rating	4.2

Question 84

Unit 1 is at 100% power.

Chemistry reports that the latest sample indicates a primary to secondary leak rate of approximately 200 gpd from the 11 Steam Generator.

- 1) Per OP O-4, Primary to Secondary Steam Generator Tube Leak Detection, once the crew commences a plant shutdown, the unit must be less than 50% in _____.
 - 2) Per LCO 3.4.13, RCS Operational Leakage, the unit must be in MODE 3 in no more than _____.
- A. 1) one hour 2) six hours
B. 1) one hour 2) ten hours
C. 1) three hours 2) six hours
D. 1) three hours 2) ten hours

Proposed Answer: A. 1) one hour 2) six hours

Explanation:

- A. Correct. Per O-4, once the downpower is started, the unit must be less than 50% in one hour. The time to reach MODE 3 is 6 hours.
- B. Incorrect. Time to less than 50% is correct, however, the time to reach MODE 3 is 6 hours. 10 hours is plausible because if its considered operational leakage, then there are 4 hours to "reduce leakage" and then 6 hours to MODE 3.
- C. Incorrect. Time is incorrect, it is one hour, not three (half of the time to MODE3). The LCO of 6 hours is correct.
- D. Incorrect. Time is incorrect, it is one hour, not three (half of the time to MODE3). The LCO of 10 hours is correct.

Technical References: LCO 3.4.13, O-4

References to be provided to applicants during exam: none

Learning Objective: 9697D Apply TS 3.4 Technical Specification LCOs.

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

Past NRC Exam

Last Two NRC Exams

Memory/Fundamental

Comprehensive/Analysis

55.43.2

X

No

No

X

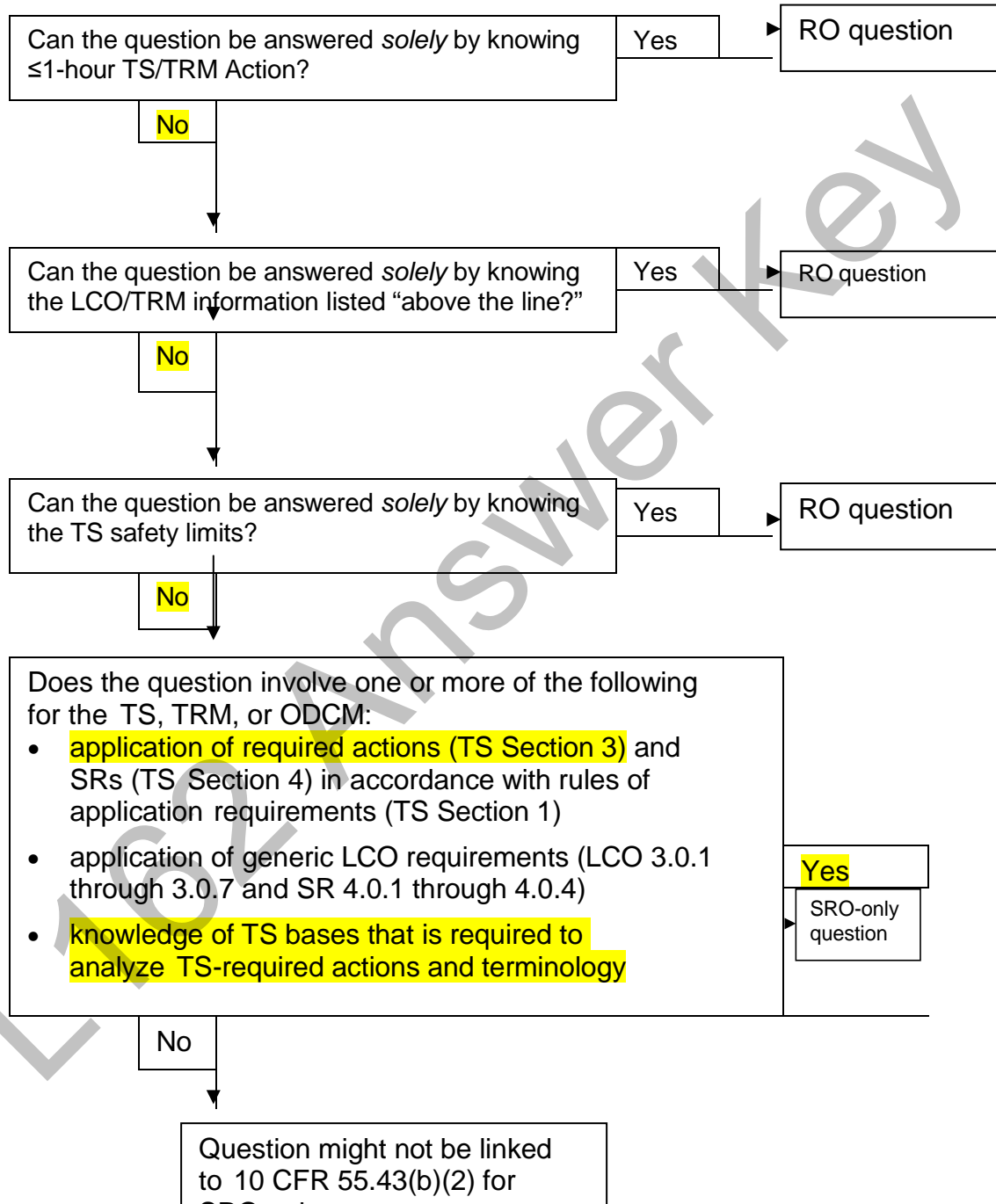
Question History:

Question Cognitive Level:

10CFR Part 55 Content:

Difficulty: 2.0

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Examination Outline Cross-Reference

E16 G2.4.2 – Containment Radiation; Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

Level	SRO
Tier #	1
Group #	2
K/A #	E16 G2.4.2
Rating	4.6

Question 85

The STA is monitoring Critical Safety Function Status trees.

If Containment Radiation monitors _____ 1) _____ exceed 12 R/HR, the crew _____ 2) _____ EOP FR-Z.3, Response To High Containment Radiation Level.

- RM-11 – Containment Air Particulate
- RM-12 – Containment Rad Gas
- RM-30, 31 – Containment High Range

- A. 1) RM-11, RM-12 2) must GO TO
- B. 1) RM-11, RM-12 2) may elect to GO TO
- C. 1) RM-30, RM-31 2) must GO TO
- D. 1) RM-30, RM-31 2) may elect to GO TO

Proposed Answer: D. 1) RM-30, RM-31 2) may elect to GO TO.

Explanation:

FR-Z.3 is a yellow path procedure, not RO knowledge, and the SRO must apply the rules of usage for FRGs to determine how the procedure is performed.

- A. Incorrect. RM-11 and RM12 are containment rad monitors but RM-30 and 31 are the rad monitors used. Transition is required for a RED or MAGENTA challenge..
- B. Incorrect. RM-11 and RM12 are containment rad monitors but RM-30 and 31 are the rad monitors used. The status tree is YELLOW and as such, it is performed at the operator's option and may be exited at any time.
- C. Incorrect. The rad monitors are RM-30 and 31 however, Z.3 is a yellow path, but unlike red or magenta challenges, performance is not mandatory.
- D. Correct. High radiation on RM-30 and 31 represents a "not satisfied" or yellow challenge to containment integrity. Because it is not a extreme or severe challenge, it is performed at the option of the operator (SFM) and may be exited at any point.

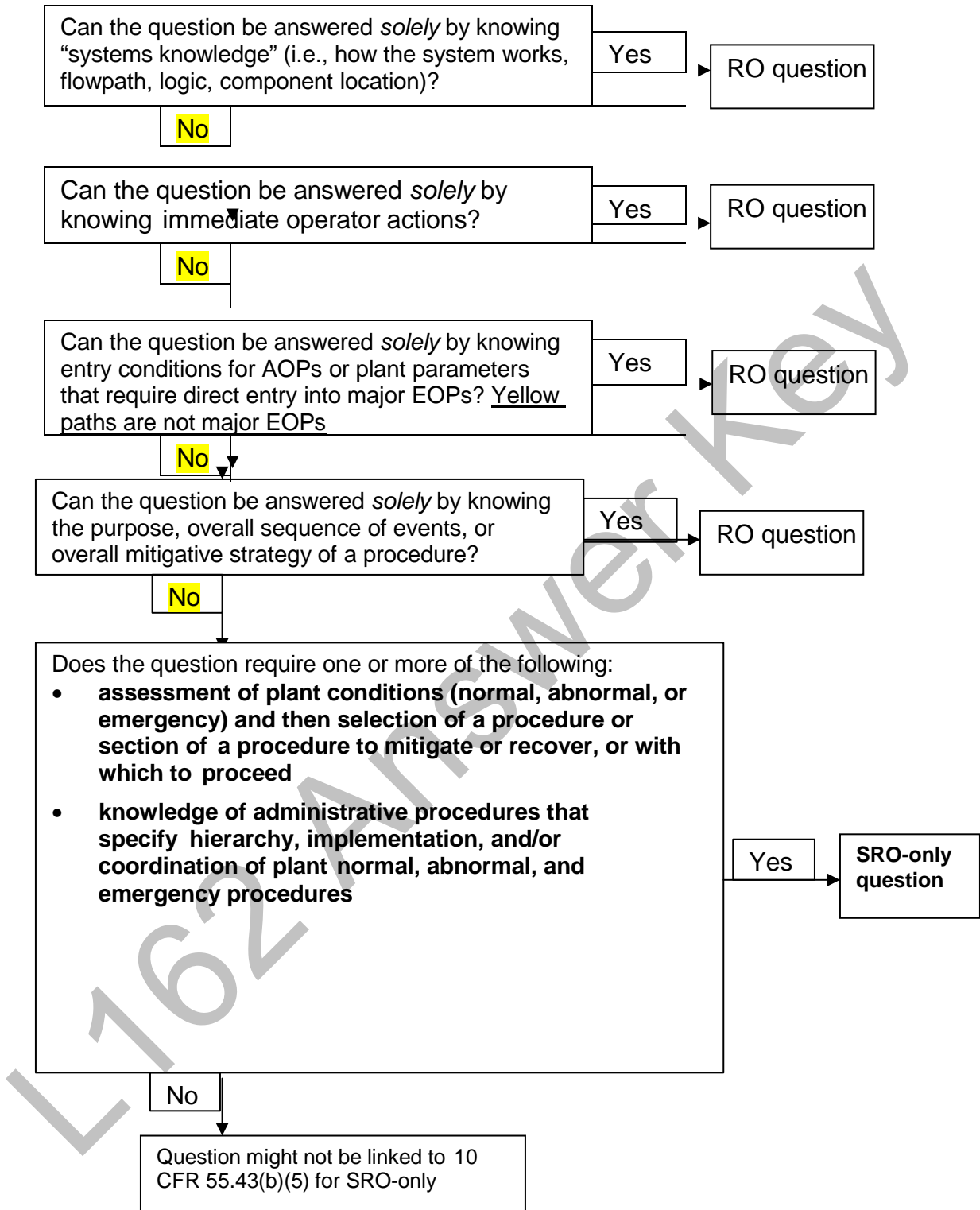
Technical References: F-0

References to be provided to applicants during exam: none

Learning Objective: 7988 Describe when procedure transitions are made while using the EOP set

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	
Difficulty: 2.0		

SRO Justification:



Examination Outline Cross-Reference

Level	SRO
Tier #	2
Group #	1
K/A #	004 A2.26
Rating	3.0

004 A2.26 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: Low VCT pressure.

Question 86

GIVEN:

- VCT level is 10% and lowering rapidly
- VCT pressure lowers rapidly to 0 psig
- Letdown flow is 75 gpm
- PK11-25, PLANT VENT RADIATION, is in alarm
- PK11-21, HIGH RADIATION, is in alarm

- 1) The Shift Foreman should enter _____ to address the leak.
 - 2) Per EP RB-2, DCCP Emergency Exposure Guidelines, if Emergency Exposure authorization is required to isolate the leak and the TSC and EOF are not activated, it should be approved by the _____.
- A. 1) OP AP-14, Tank Ruptures 2) Shift Manager
- B. 1) OP AP-14, Tank Ruptures 2) Emergency Director
- C. 1) OP AP-17, Loss of Charging 2) Shift Manager
- D. 1) OP AP-17, Loss of Charging 2) Emergency Director

Proposed Answer: A. 1) OP AP-14, Tank Ruptures 2) Shift Manager

Explanation:

- A. Correct. Low VCT pressure and level are indicative of a VCT rupture. The procedure to address it is OP AP-14. In AP-14, step 4 is to implement radiological emergency procedures. First bullet is RB-2. RB-2 states the SM or the SEC or ED approves emergency exposures.
- B. Incorrect. The procedure is AP-14, however, the SM approves emergency exposures prior to EOF/TSC activation. Plausible as the ED is in charge once the EOF is activated.
- C. Incorrect. PK11-21 could be in alarm for either loss of charging or tank rupture. Also, VCT level would lower for a large charging line leak (but pressure would not fall to 0 psig). The SEC approves emergency exposure.
- D. Incorrect. The appropriate procedure is AP-14 not AP-17 and the SM approves emergency exposures.

Technical References: AR PK11-21, AR PK11-25, OP AP-14, AP-17, RB-2

References to be provided to applicants during exam: none

Learning Objective: 3477P Given an abnormal condition, summarize the major actions of OP AP-14 to mitigate an event in progress.

9848 - State who may authorize emergency doses

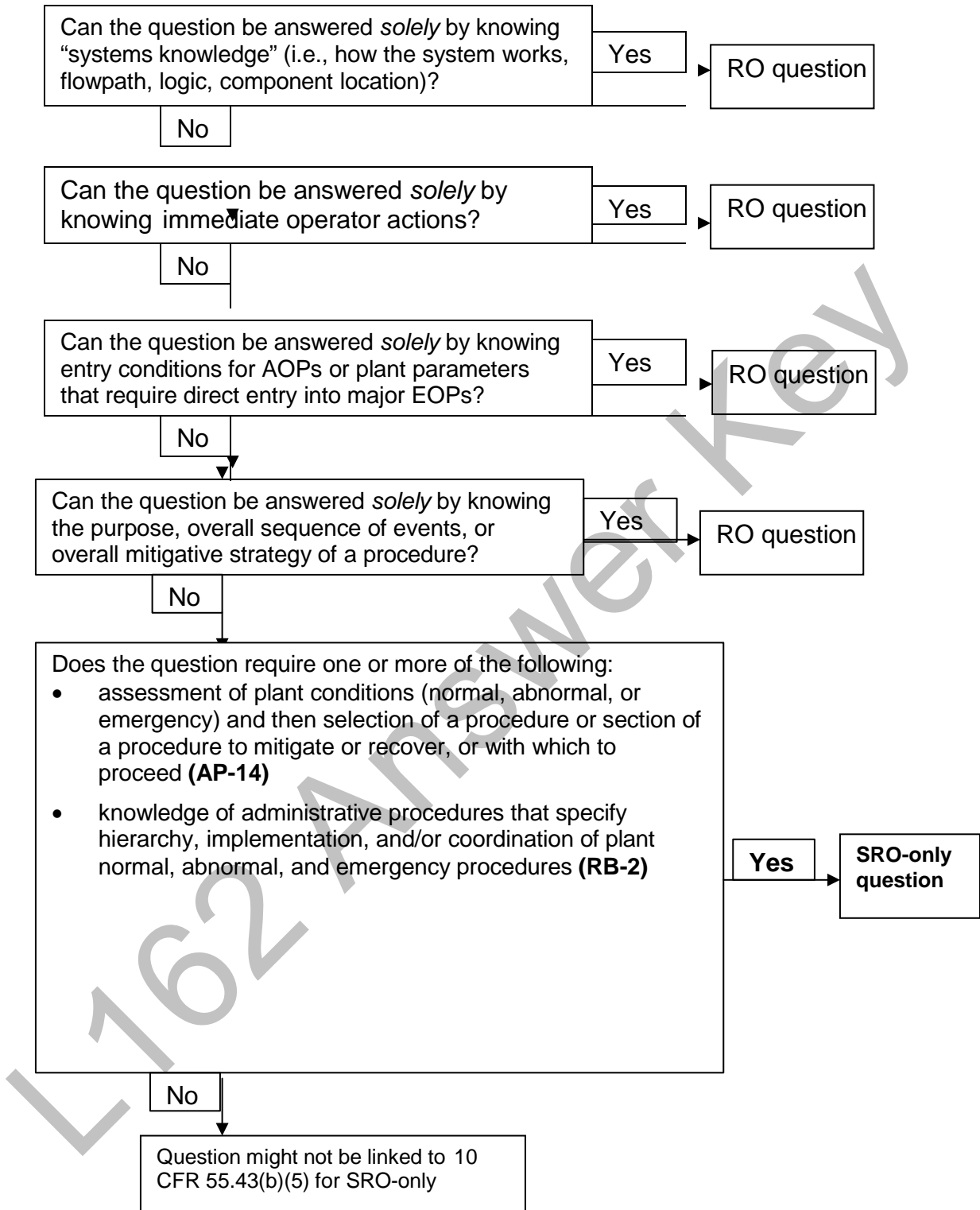
Question Source: Bank #
(note changes; attach parent) Modified Bank #
New

X

Question History:	Past NRC Exam	No
	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.4	
Difficulty: 3.0		

L162 Answer Key

SRO Justification:



Examination Outline Cross-Reference

007 A2.04 Ability to (a) predict the impacts of the following malfunctions or operations on the PTRS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Overpressurization of the waste gas vent header.

Level	SRO
Tier #	2
Group #	1
K/A #	007 A2.04
Rating	2.9

Question 87

GIVEN:

- Unit 1 is at 100% power
- PK05-20, PZR Relief/Safety Valve Open – ON
- PK05-23, PZR Safety or Relief Line Temp – ON
- Sonic Flow indication is 0.001
- PRT pressure is 15 psig and rising
- Pressurizer pressure dropped rapidly

The Shift Foreman is addressing PK05-20.

In accordance with AR PK05-20, the Shift Foreman should enter _____. Additionally, _____, if open, will close.

- A. AP-1, Excessive Reactor Coolant System Leakage; RCS-8034A/B, Gas Analyzer valves
- B. AP-1, Excessive Reactor Coolant System Leakage; PCV-472, PRT Vent Header Pressure Control valve
- C. AP-13, Malfunction of Reactor Pressure Control; RCS-8034A/B, Gas Analyzer valves
- D. AP-13, Malfunction of Reactor Pressure Control; PCV-472, PRT Vent Header Pressure Control valve

Proposed Answer: D. AP-13, Malfunction of Reactor Pressure Control; PCV-472, PRT Vent Header Pressure Control valve

Explanation:

The SRO will evaluate the plant conditions and determine which of the two abnormal procedures adequately address the plant conditions. AP-1 would eventually address the open PORV/Safety however, AP-13 will address the problem immediately (and is specified in the PK)

- A. Incorrect. While RCS is leaking, it is due to an open PORV/Safety (no sonic flow). The 8034 valves do have an auto close, but its phase A, not high PRT pressure.
- B. Incorrect. While RCS is leaking, it is due to an open PORV/Safety. The PK directs the operator to OP AP-13. At 10 psig of PRT pressure, PCV-472 valve closes isolating the PRT from the vent gas header.
- C. Incorrect. The procedure to be entered is AP-13. The 8034 valves do have an auto close, but its phase A, not high PRT pressure.
- D. Correct. While RCS is leaking, it is due to an open PORV/Safety. The PK directs the operator to OP AP-13. At 10 psig of PRT pressure, PCV-472 valve closes isolating the PRT from the vent gas header.

Technical References: AP-1, AP-13, AR PK05-20, AR PK05-23

References to be provided to applicants during exam: none

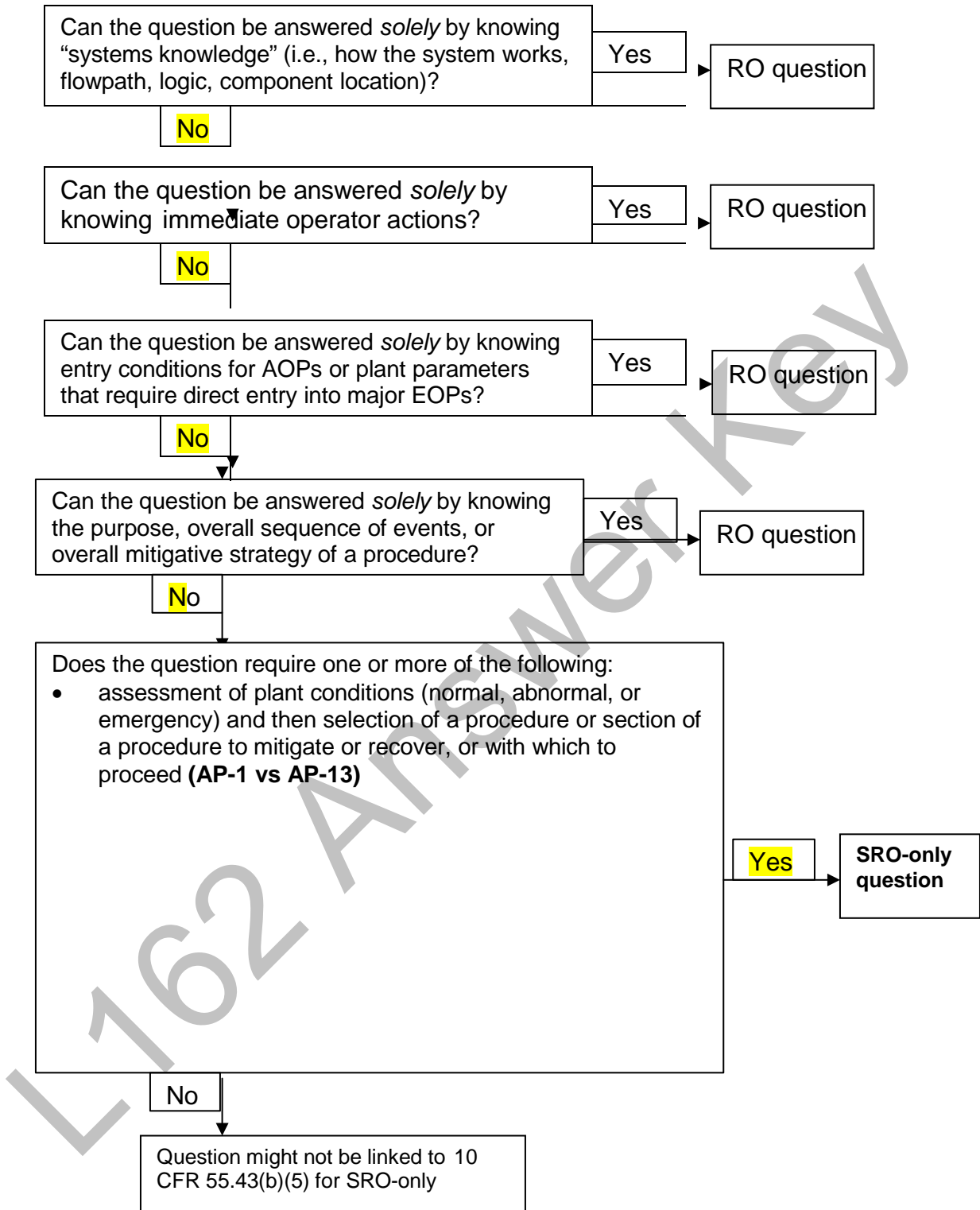
Learning Objective: 3478 - Given initial conditions, assumptions, and symptoms, determine the correct abnormal operating procedure to be used to mitigate an operational event

Question Source:	Bank #88 DCPD NRC 11/2012	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Past NRC Exam	Yes
	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 3.5

L162 Answer Key

SRO Justification:



Examination Outline Cross-Reference**022 G2.4.41 – Containment cooling: Knowledge of the emergency action level thresholds and classifications.**

Level	SRO
Tier #	2
Group #	1
K/A #	022 G2.4.41
Rating	4.6

Question 88

According to the EAL, Table F-1, Fission Product Barrier Matrix, which of the following is the minimum depressurization equipment required to prevent a potential loss of the Containment barrier?

	<u>Containment Spray Pump(s)</u>	<u>CFCU(s)</u>
A.	1	2
B.	1	1
C.	2	2
D.	2	1

Proposed Answer: A. 1 2

Explanation:

During an accident, the containment is cooled and depressurized by using containment spray and CFCUs.

- A. Correct. According to a note on Table 1, one containment spray pump and two CFCUs constitute a train of depressurization equipment and is the MINIMUM required.
- B. Incorrect. Only one containment spray pump is required. However, 2 CFCUs are required. Plausible – EAL may require less to make a classification.
- C. Incorrect. Only one CS pump is required. This is more than the minimum required. Plausible because less than 2 requires LCO entry. The number of CFCUs required is 2.
- D. Incorrect. This is more containment spray pumps than what is required. Number of CFCUs is less than required.

Technical References: EAL Fission Barrier Table, E-Plan Appendix D Bases, LCO 3.6.6**References to be provided to applicants during exam:** none

Learning Objective: Given an emergency classification or approval of a protective action recommendation (PAR), use EP G-3 to complete a DCPD Emergency Notification Form with 100% accuracy within 15 minutes. (42288)

Question Source:

(note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
	Past NRC Exam	No

Question History:

Last Two NRC Exams	No
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Question Cognitive Level:

Memory/Fundamental	X
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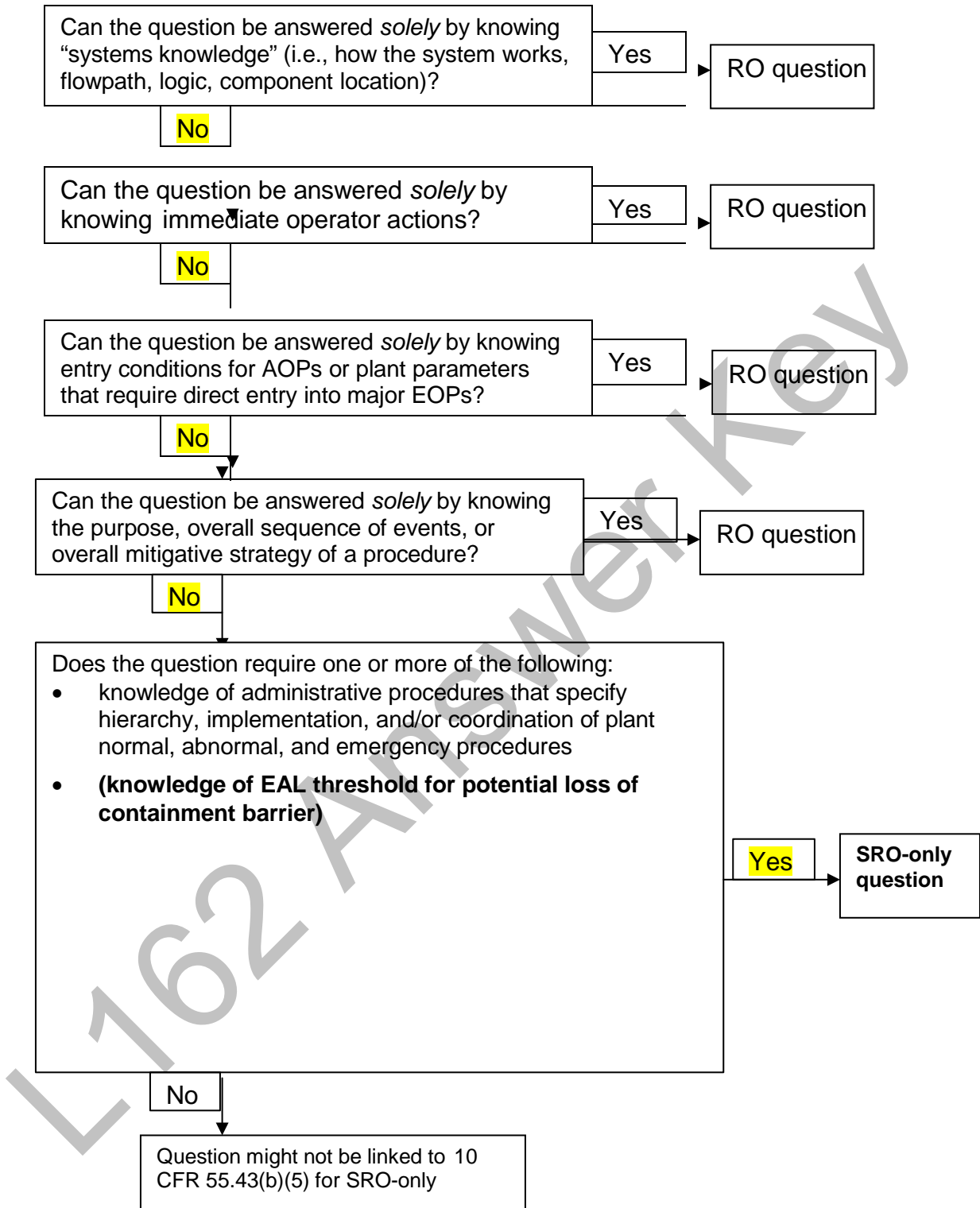
Comprehensive/Analysis	
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10CFR Part 55 Content:

55.43.1	
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Difficulty: 2.5

SRO Justification:



Examination Outline Cross-Reference**073 G2.1.32 Process Radiation Monitoring: Ability to explain and apply system limits and precautions.**

Level	SRO
Tier #	2
Group #	2
K/A #	073 G2.1.32
Rating	4.0

Question 89

What is the reason for limiting the open time of the containment purge valves to no more than 200 hours for the year?

- A. To prevent exceeding the NPDES permit.
- B. To prevent valve erosion and subsequent excessive leakage past the valves when they are closed.
- C. To minimize the probability of a LOCA occurring while the valves are open, limiting the offsite boundary doses.
- D. To minimize the probability the valves will not fully close when the purge is secured, resulting in a breach of Containment integrity.

Proposed Answer: C. To minimize the probability of a LOCA occurring while the valves are open, limiting the offsite boundary doses.

Explanation:

- A. Incorrect. The NPDES covers releases to the environment, but releases to the ocean, not from Containment.
- B. Incorrect. These butterfly valves are not fully open (limited to 50 degrees of travel), so its plausible that there is concern of wear.
- C. Correct .Containment purges rely on RM-44A and 44B to be OPERABLE. These are process rad monitors. Their operation is assumed to close the purge valves that are opened. Per ECG 23.3, The purging time restriction is meant to minimize the probability of a LOCA while conducting purging operations and thereby limit offsite boundary doses.
- D. Incorrect. The valves are limited in travel because these valves have not been qualified to close under accident conditions. Knowing that there is a concern with closing makes this distractor plausible.

Technical References: CAP A-6A, ECG 23.3.

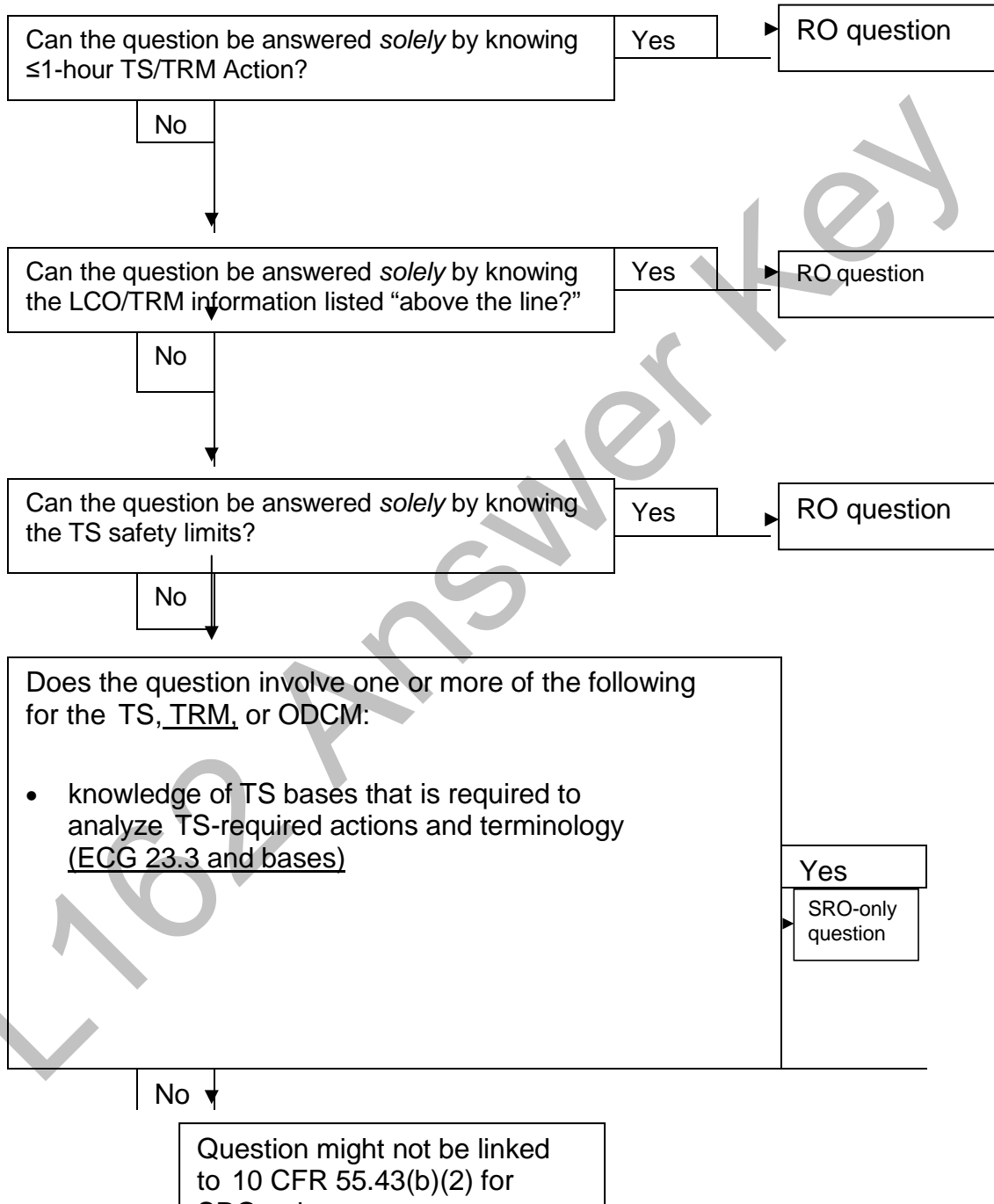
References to be provided to applicants during exam: none

Learning Objective: 7428 - State gaseous radwaste system administrative controls

Question Source:	Bank #91 DCPN NRC 10/2016	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	Yes
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.2	

Difficulty: 3.0

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Examination Outline Cross-Reference

Level	SRO
Tier #	2
Group #	1
K/A #	076 G2.2.25
Rating	4.2

076 G2.2.25 Service Water - Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Question 90

According to the bases for LCO 3.7.8, Auxiliary Saltwater (ASW) System, a train of ASW requires ___1)___ vacuum relief valve(s) to be OPERABLE. The pump vault drain check valve ___2)___ required for the train to be OPERABLE.

- A. 1) at least one 2) is NOT
- B. 1) both 2) is NOT
- C. 1) at least one 2) is
- D. 1) both 2) is

Proposed Answer: C. 1) at least one 2) is

Explanation:

An ASW train is considered OPERABLE during MODES 1, 2, 3, and 4 when:

- a. The pump is OPERABLE; and
- b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the PG&E Design Class I function are OPERABLE. This requires that at least one vacuum relief valve be OPERABLE
- c. The associated pump vault drain check valve is OPERABLE. The ASW pump vault check valves prevent flooding of the ASW pump vaults during design flood events.

- A. Incorrect. First part is correct, failure of only one relief valve does not make the train inoperable. Failure of the vault pump check valve makes the train inoperable. Plausible if the candidate knows the relief valves are required and believes that both are required (like both ASW pumps are required for the LCO to be satisfied) and that a room (vault) pump valve is not required.
- B. Incorrect. At least one is required, both is more than what is required. Plausible if the bases is not known and its felt a pump vault check valve is not covered by Tech Specs.
- C. Correct. Only ONE is required and the check valve is also required per the bases.
- D. Incorrect. ONLY ONE is required. Plausible if the candidate knows the relief valves are required and believes that both are required (like both ASW pumps are required for the LCO to be satisfied). Second part is correct, the check valve is also required.

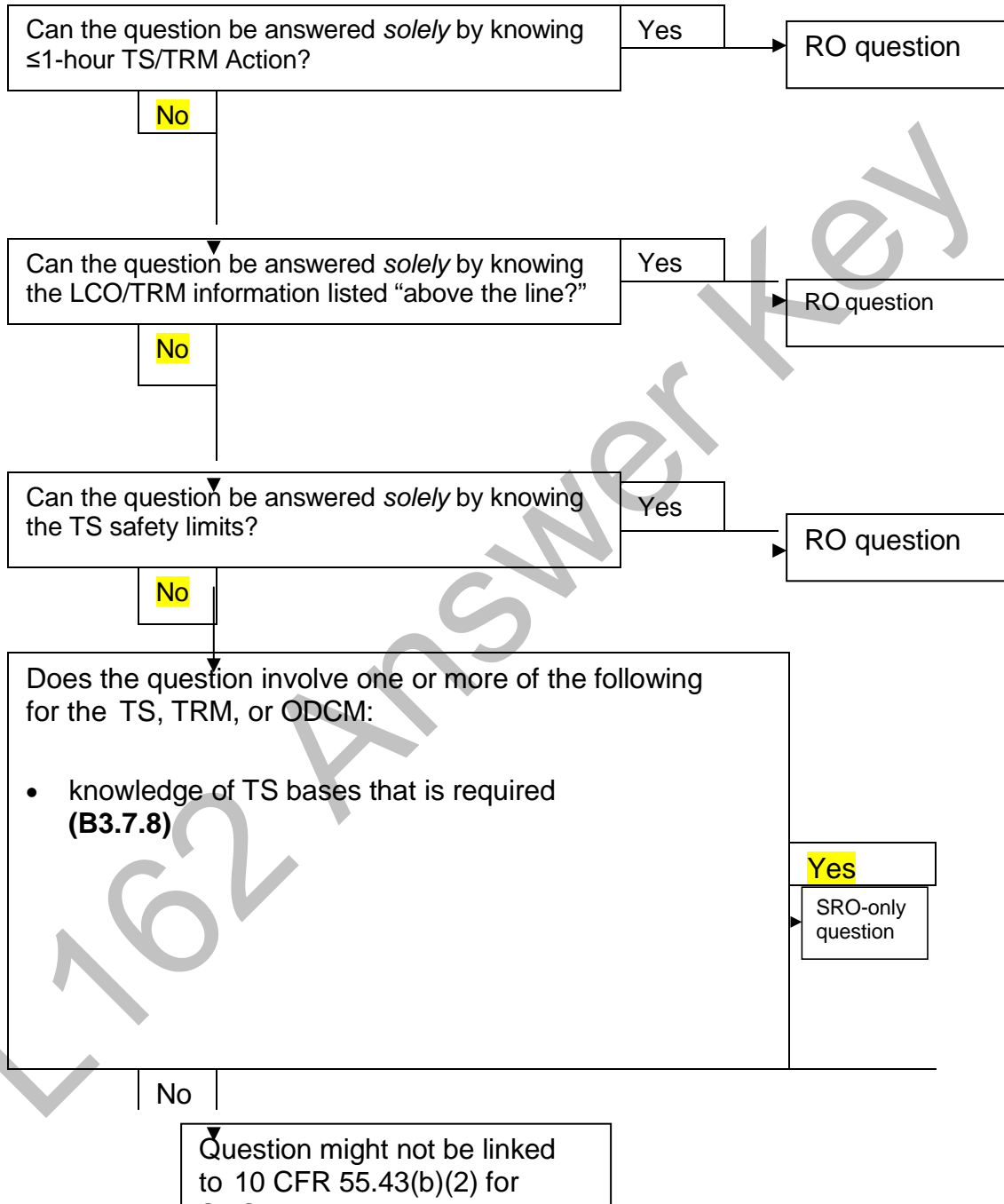
Technical References: B 3.7.8

References to be provided to applicants during exam: none

Learning Objective: 9694G Apply TS 3.7 Technical Specification bases

Question Source:	Bank #90 DCPD NRC Exam 03/2012	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.2	
Difficulty: 3.0		

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Examination Outline Cross-Reference

016 A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure.

Level	SRO
Tier #	2
Group #	2
K/A #	016 A2.01
Rating	3.1

Question 91

GIVEN:

- Unit 1 is at 100% power
- Tave is 572°F
- Pressurizer level channels:
 - LT-459 indicates 60%
 - LT-460 indicates 51%
 - LT-461 indicates 56%
- Pressurizer pressure channels
 - PT-455 indicates 2235 psig
 - PT-456 indicates 2180 psig
 - PT-457 indicates 2215 psig
- RM-3, Oily Water Separator Detector, analog meter needle is on the low limit peg and motionless

Based on these indications, the Shift Foreman should declare _____ inoperable.

1. A pressurizer level channel
 2. A pressurizer pressure channel
 3. RM-3
- A. 1 ONLY
B. 1 and 3
C. 2 ONLY
D. 2 and 3

Proposed Answer: B. 1 and 3

Explanation:

NOTE: CHANNEL CHECKS are required to satisfy the following surveillance requirements
RM-3 – ECG 39.3, SR 39.3.1

Pressurizer level and pressure – LCO 3.3.1, SR 3.3.1.1

- A. Incorrect. Pressurizer level is greater than the allowable variance of 8.5%. Pressure is within tolerance, however, RM-3 is inoperable.
- B. Correct. level is outside of tolerance (greater than 8.5%) and RM-3 is inoperable (needle should have a slight oscillation).
- C. Incorrect. Although there is a noticeable difference in pressure channels, they are within 5% (62.5 psig).
- D. Incorrect. Pressure channels are operable, RM-3 is not.

Technical References: Ops policy B-5, LCO 3.3.1, ECG 39.3

References to be provided to applicants during exam: none

Learning Objective:

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #95 DCP L111 NRC 11/2012

New

Past NRC Exam

Yes

Question History:

Last Two NRC Exams

No

Question Cognitive Level:

Memory/Fundamental

Comprehensive/Analysis

X

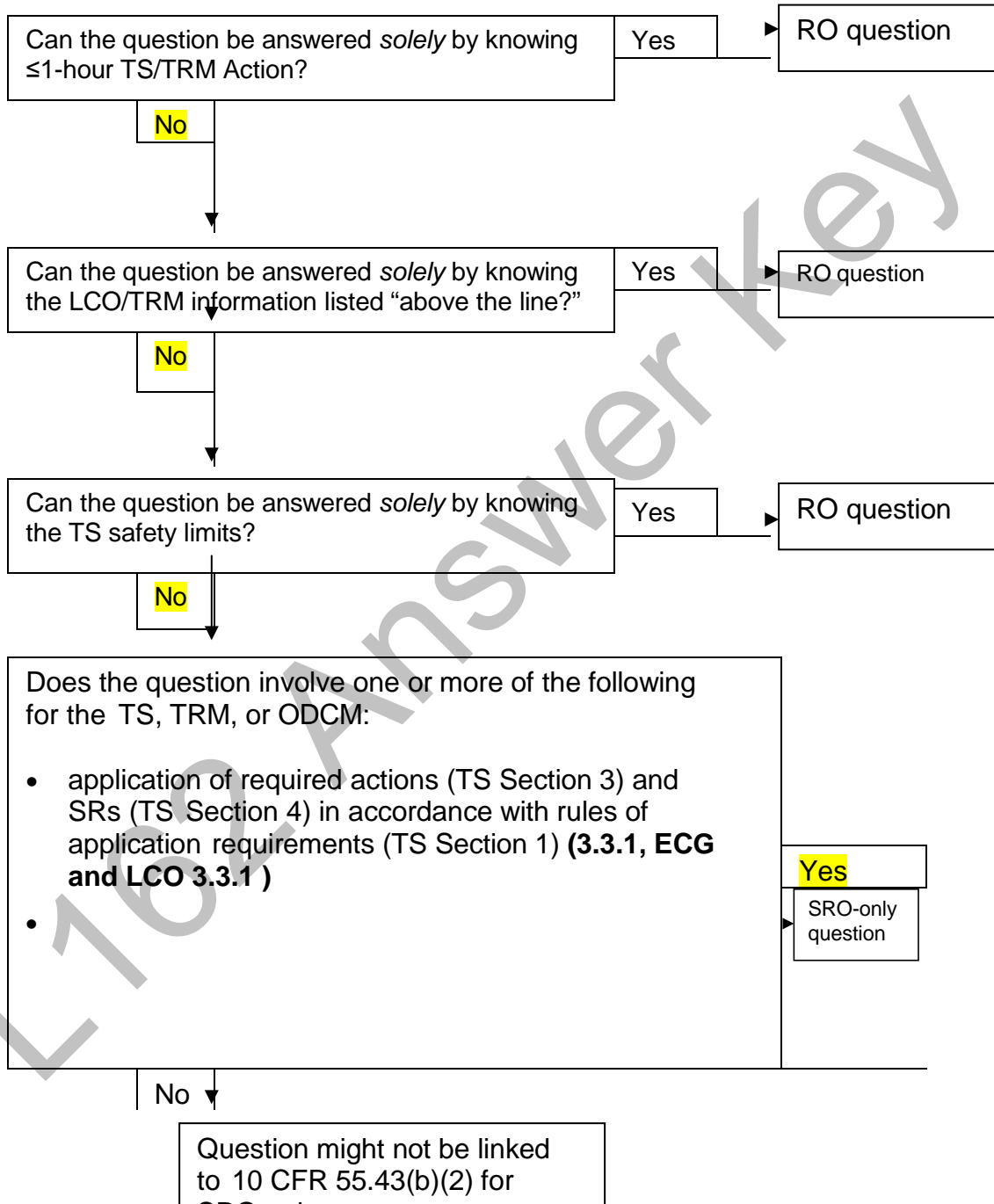
10CFR Part 55 Content:

55.43.2

Difficulty: 3.0

L162 Answer Key

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Examination Outline Cross-Reference

Level	SRO
Tier #	2
Group #	2
K/A #	035 A2.05
Rating	3.4

035 A2.05 Ability to (a) predict the impacts of the following malfunctions or operations on the S/G; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Unbalanced flows to the S/Gs.

Question 92

GIVEN:

- A natural circulation cooldown is in progress on Unit 2 at 35°F/hour in accordance with EOP E-0.2, Natural Circulation Cooldown
- Pressurizer level is 35% and stable
- RVLIS is 100%
- Steam Generator 21 cannot be fed or steamed
- ΔT in all RCS loops is 25°F

RCS loop 21 ΔT has lowered to 10°F and T_{hot} has stabilized over the last 20 minutes.

- 1) In accordance with EOP E-0.2, RCS loop 21 is becoming _____.
 - 2) The Shift Foreman will direct the crew to _____.
- A. 1) stagnant 2) raise the cooldown rate
 B. 1) stagnant 2) lower the cooldown rate
 C. 1) voided 2) raise the cooldown rate
 D. 1) voided 2) lower the cooldown rate

Proposed Answer: B. 1) stagnant 2) lower the cooldown rate

Explanation:

- A. Incorrect. The consequence is correct. However, the cooldown rate needs to be lowered. Plausible because it could be that its believed increasing the cooldown will lower T_{cold} , which could lower the loop 21 T_{cold} , possibly restarting cooling in the loop.
- B. Correct. Step 10 defines an inactive loop, is a loop that cannot be fed or steamed. Step 11 states that if hot legs in the inactive loop are not lowering at the same rate of the other loops, it is stagnant. If its stagnant (T_{hot} stable) then the RNO states to lower the cooldown rate (by a factor of 2).
- C. Incorrect. Voids is plausible as this is also monitored during E-0.2 and is a potential consequence of too high a cooldown rate. Cooldown rate is lowered not raised.
- D. Incorrect. Voids is plausible as this is also monitored during E-0.2 and is a potential consequence of too high a cooldown rate. Action to lower cooldown rate is correct.

Technical References: E-0.2 step 10 and 11and background

References to be provided to applicants during exam: None

Learning Objective: Explain the operator actions that can initiate or enhance natural circulation. (8905)

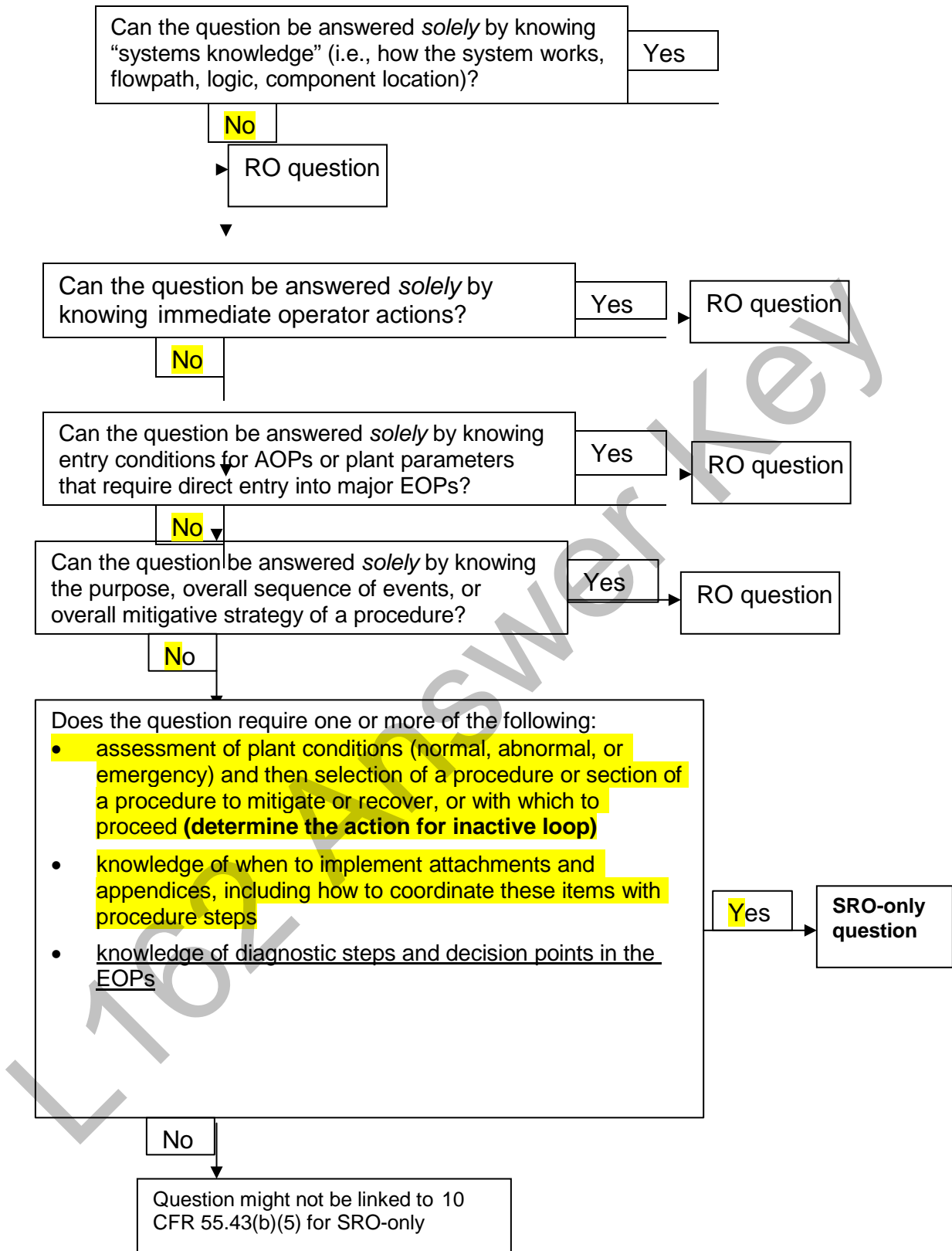
Explain the basis for emergency procedure steps. (7920C)

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	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	
Difficulty: 3.0		

L162 Answer Key

SRO Justification:



Examination Outline Cross-Reference

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

Level	SRO
Tier #	2
Group #	2
K/A #	068 A2.04
Rating	3.3

Question 93

GIVEN:

- Unit 1 is at 100% power
 - A High Radiation alarm is received for RE-23, Steam Generator Blowdown Effluent Detector
 - The auto actions due to the high radiation signal do not occur but are performed by the operator within approximately five minutes of receiving the alarm
- 1) Which of the following identifies an automatic action that should have occurred?
- 2) According to the bases for ECG 39.3, Radioactive Liquid Effluent Monitoring Instrumentation, the automatic isolation setpoint is designed to ensure that isolation occurs prior to exceeding the limits of:
- A. 1) Blowdown isolation valves, Inside Containment, FCV-760, 761, 762, and 763 close.
2) 10 CFR Part 20, Standards For Protection Against Radiation.
- B. 1) Blowdown isolation valves, Inside Containment, FCV-760, 761, 762, and 763 close.
2) the state discharge permit.
- C. 1) Blowdown isolation valves, Outside Containment, FCV-151, 154, 157, and 160, close.
2) 10 CFR Part 20, Standards For Protection Against Radiation.
- D. 1) Blowdown isolation valves, Outside Containment, FCV-151, 154, 157, and 160, close.
2) the state discharge permit.

Proposed Answer:

- C. 1) Blowdown isolation valves, Outside Containment, FCV-151, 154, 157, and 160, close.
2) 10 CFR Part 20, Standards For Protection Against Radiation.

Explanation:

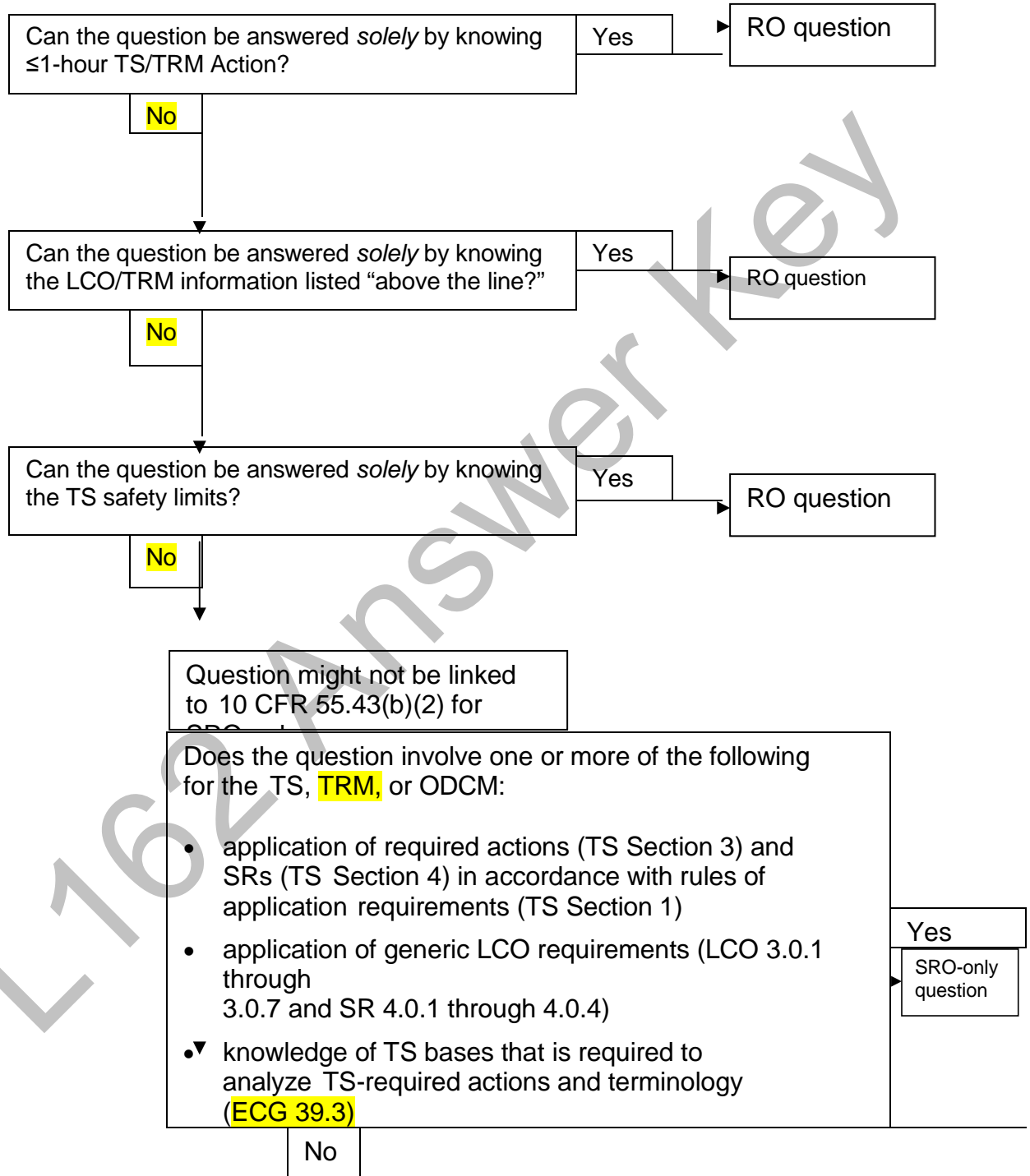
- A. Incorrect. 10 CFR 20 is correct. The outside, not inside isolation valves close.
- B. Incorrect. The state discharge permit is a limit the plant operates by but it is not what the ECG limit is based upon.
- C. Correct. The outside isolation valves close and as stated in the bases for ECG 39.3, operable, releases may occur if the actions stated occur. ECG 39.3 bases states: 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.
- D. Incorrect. The outside isolation valves close but it is not the state permit the ECG is designed to meet but the 10 CFR part 20.

Technical References: ECG-39.3, LD-2**References to be provided to applicants during exam:** none**Learning Objective:** 66068 - Apply the requirements of System 39 ECGs.

Question Source: (note changes; attach parent)	Bank #93 DCPD NRC Exam 08/14 Modified Bank # New Past NRC Exam	X Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 X
10CFR Part 55 Content:	55.43.1	
Difficulty: 2.5		

L162 Answer Key

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Examination Outline Cross-Reference

Level	SRO
Tier #	3
Group #	1
K/A #	G2.1.26
Rating	3.6

G2.1.26 Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen).

Question 94

A hazardous material emergency occurs in the U1 Containment due to a leak that is determined to be Immediately Dangerous to Life and Health (IDLH).

- 1) Per CP M-9A, Hazardous Materials Incident – Initial Emergency Response/Mitigation Procedure, the _____ is/are contacted for immediate assistance.
- 2) Due to the IDLH environment, the maximum time to notify the NRC is _____ hour(s).
 - A. 1) DCPD Fire Department 2) one
 - B. 1) DCPD Fire Department 2) four
 - C. 1) Chemistry and Environmental Operations Departments 2) one
 - D. 1) Chemistry and Environmental Operations Departments 2) four

Proposed Answer: A.1) DCPD Fire Department 2) one

Explanation:

DCPD has had 2 emergency ALERT classifications due to IDLH, deficient oxygen concentration in containment and CARDOX actuation in the Turbine Building.

- A. Correct. Per the procedure, the fire department is called. Due to IDLH, an ALERT emergency plan classification will be made and the NRC must be notified within one hour.
- B. Incorrect. While fire department is correct, the NRC must be notified within one hour. Plausible if the IDLH is not put together with the emergency plan. Then either it could be thought that there is no reason to notify the NRC or that there required notification time is longer, like four or eight hours.
- C. Incorrect. The fire department, not the C and EO department is called. The NRC is notified. Chemistry is plausible as it seems logical if there is this type of issue and its not known that the fire department is contacted.
- D. Incorrect. The fire department is notified and NRC notification is required within one hour due to activation of the emergency plan.

Technical References: CP-M-9A and EAL chart (HA 3.1), DCPD OE SAPN 50934898

References to be provided to applicants during exam: none

Learning Objective: 42159 Determine reporting requirements of 24 hours or less

Question Source:

(note changes; attach parent)

Bank #	
Modified Bank #	
New	X
Past NRC Exam	No
Last Two NRC Exams	No
Memory/Fundamental	
Comprehensive/Analysis	X

Question History:

Question Cognitive Level:

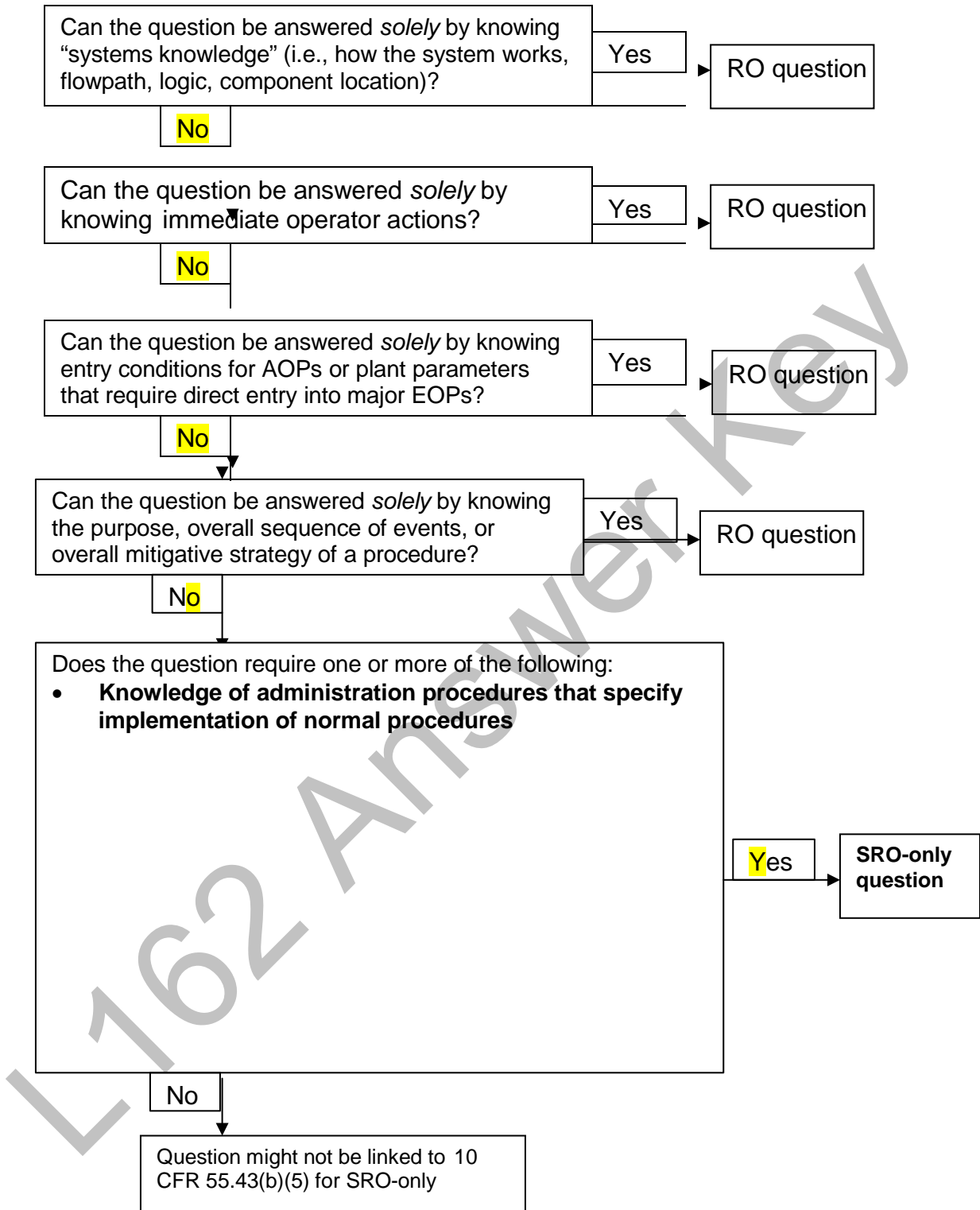
10CFR Part 55 Content:

55.43.3

Difficulty: 2.5

L162 Answer Key

SRO Justification:



Examination Outline Cross-Reference

Level	SRO
Tier #	3
Group #	1
K/A #	G2.1.35
Rating	3.9

G2.1.35 -Knowledge of the fuel-handling responsibilities of SROs**Question 95**

In accordance with OP B-8DS1, Core Unloading, which of the following defueling activities is the first evolution performed requiring the direct supervision of the Refueling SRO in Containment because it is a CORE ALTERATION?

- A. Unlatching RCCAs
- B. Lifting the upper internals
- C. Moving the first fuel assembly
- D. Lifting the reactor vessel head

Proposed Answer: A. Unlatching RCCAs

Explanation:

- A. Correct. Per OP B-8DS1, the Refueling SRO is responsible for supervising Core Alterations. RCCA unlatching and moving of fuel assemblies are core alterations, of the two, RCCA unlatching would be the first one performed.
- B. Incorrect. Lifting the internals, is in the vessel but is not a **fuel, sources, or reactivity control components**, within the reactor vessel.
- C. Incorrect. This is a core alteration, but comes after unlatching.
- D. Incorrect. Performed prior to unlatch, but is not a core alteration **Technical**

References: OP L-6, OP B-8DS1

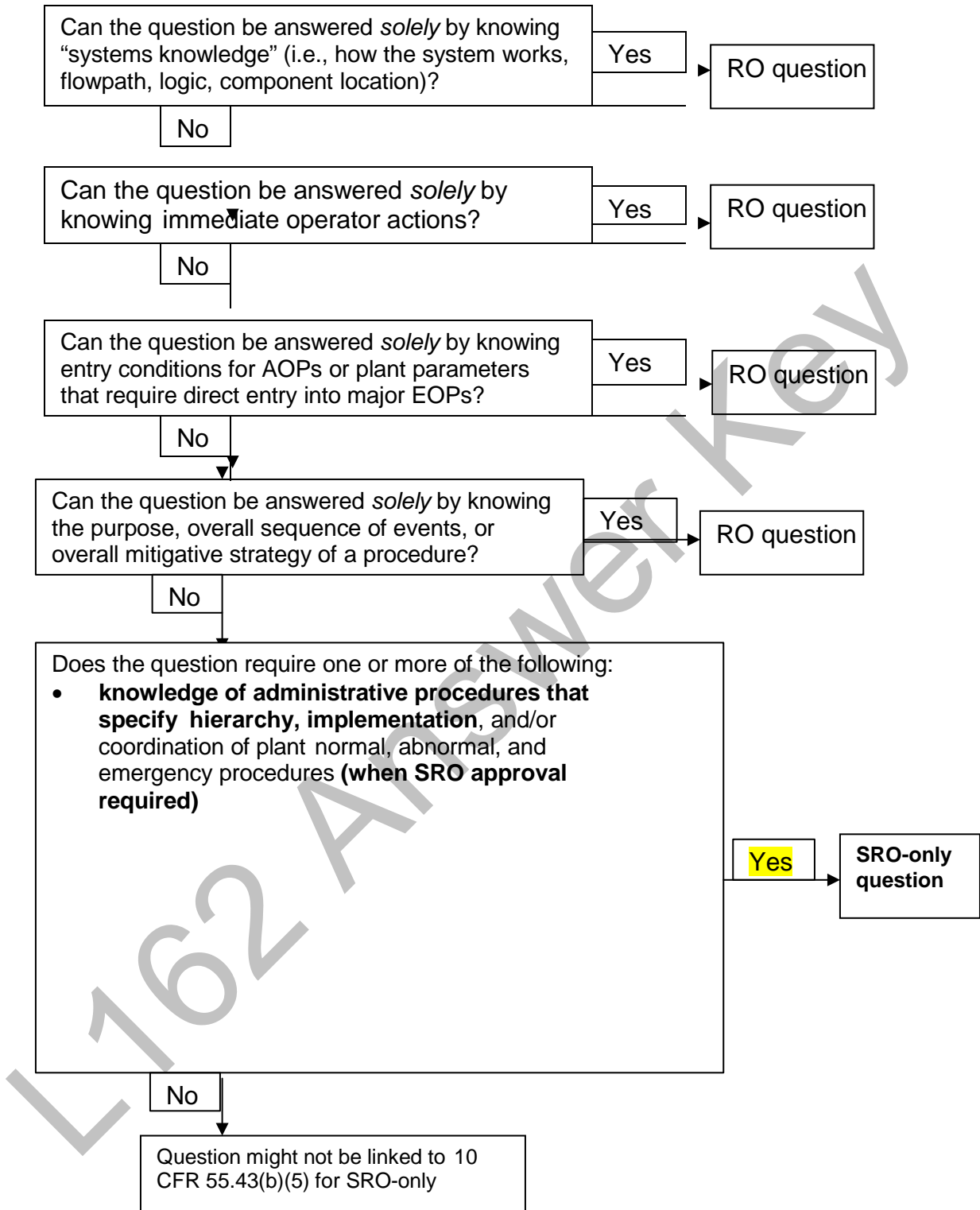
References to be provided to applicants during exam: none

Learning Objective:6497 - State the responsibilities and duties of Refueling SRO

Question Source:	Bank #95 DCPN NRC 10/2016	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Past NRC Exam	Yes
	Last Two NRC Exams	Yes
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.7	

Difficulty: 2.0

SRO Justification:



Examination Outline Cross-Reference

Level	SRO
Tier #	3
Group #	2
K/A #	G2.2.6
Rating	3.6

G2.2.6 - Knowledge of the process for making changes to procedures.

Question 96

A plant procedure which affects plant operations is being performed.

Per AD2.ID1, Procedure and Work Plan Use and Adherence, a NON-CONDITIONAL N/A in the procedure being performed _____.

1. may be marked N/A without explanation
 2. is marked N/A with the reason for the N/A documented in the remarks or cover sheet
 3. requires SRO approval
- A. 1 ONLY
- B. 2 ONLY
- C. 3 ONLY
- D. 2 and 3

Proposed Answer: D. 2 and 3

Explanation:

- A. Incorrect. Per AD2.ID1, conditional N/A are typically IF statements or provide specific conditions for being marked N/A and maybe marked N/A without explanation. Non conditional N/A are steps without conditional statements and require:
1. Write N/A at end of step. Perform one of the following to indicate that a consecutive group of steps is N/A:
 - Write N/A at the end of the first step, and draw a vertical line through the remaining steps.
 - Block-out the steps not to be performed. Write N/A in this block.
 2. Document the reason for the N/A in the document remarks or cover sheet. Initial and date this explanation.
 3. A technically cognizant supervisor (Senior Reactor Operator if the procedure affects plant operations) shall perform the following:
 - Ensure that the explanation for the N/A is appropriate.
 - Document approval by also initialing and dating the explanation.
- B. Incorrect. The reason must be documented, however the SRO must approve the N/A.
- C. Incorrect. SRO approval is required, however, documentation is required..
- D. Correct. Both 2 and 3 are required.

Technical References: step 5.7, AD2.ID1

References to be provided to applicants during exam: none

Learning Objective: 9809 - State the standard for procedure compliance

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

Question Cognitive Level:

Memory/Fundamental
Comprehensive/Analysis
55.43.3

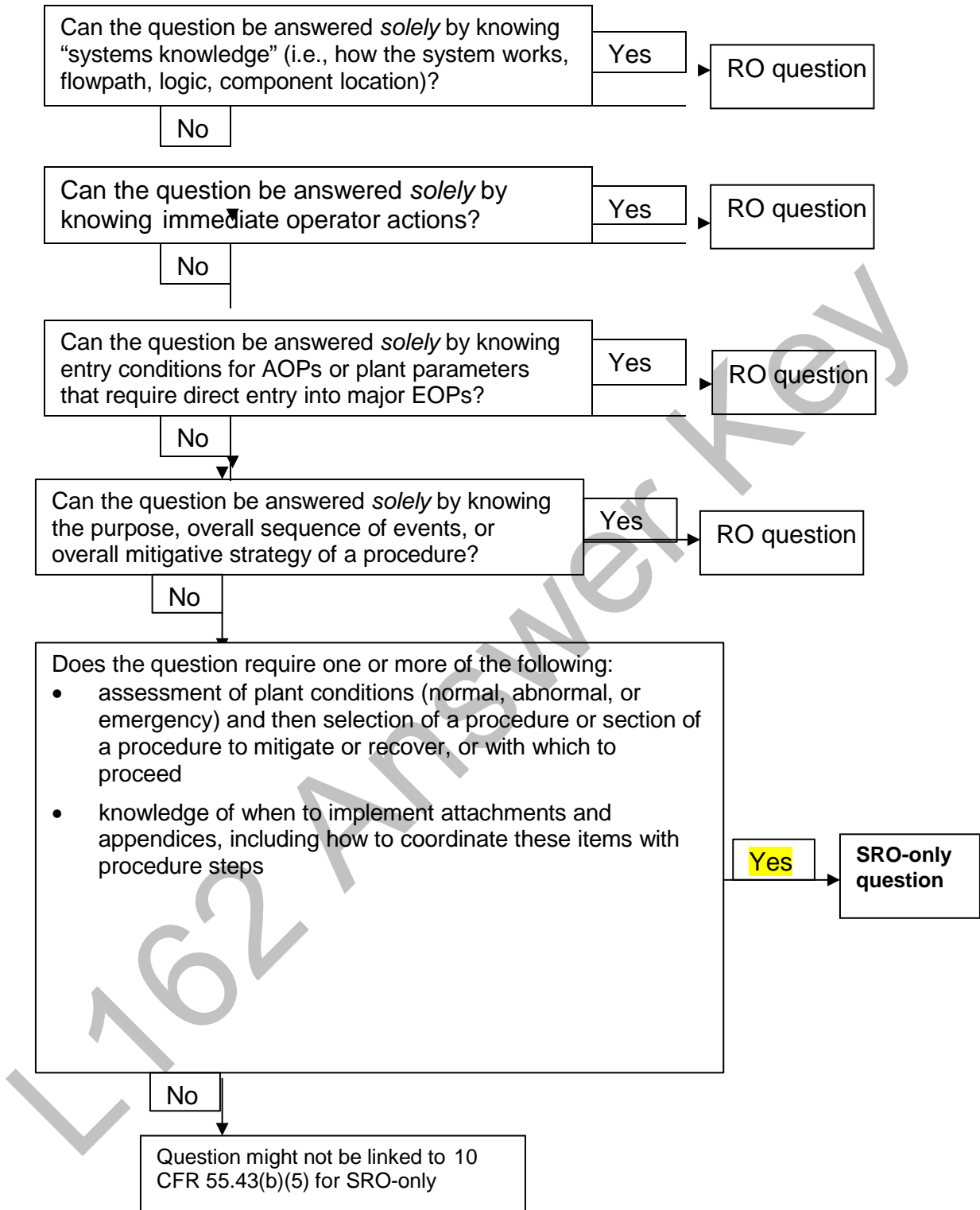
X

10CFR Part 55 Content:

Difficulty: 2.0

L162 Answer Key

SRO Justification:



Examination Outline Cross-Reference

Level	SRO
Tier #	3
Group #	2
K/A #	G2.2.17
Rating	3.8

G2.2.17 Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.

Question 97

Unit 1 is at 100% power.

Unit 1 is planning to commence a ramp to 75% power for emergent maintenance.

Per Operations Policy B-1, Communications With Generation And Transmission Organizations, the _____ has the responsibility to notify the Grid Control Center (GCC).

- A. Shift Manager
- B. Unit 1 Shift Foreman
- C. Duty team Station Director
- D. Work Control Shift Foreman

Proposed Answer: A. Shift Manager

Explanation:

- A. Correct. Per Policy B-1, load changes greater than 10 MWe require notifying STES and GCC. For an unplanned ramp, the SM is responsible for making the notification.
- B. Incorrect. While Unit 1 is the affected unit, the SM not the SFM is responsible to make the notification.
- C. Incorrect. The duty team SD leads the duty team which "provides a balanced leadership operational oversight model on non-workdays, as well as when emergent leadership needs arise after hours during the work week".
- D. Incorrect. The WCSFM will approve the package for the work, but the notification of GCC and STES is not part of that approval. The SM makes the notification.

Technical References: OP Policy B-1 step 5 – notifications, AD8.DC59, Forced Outage

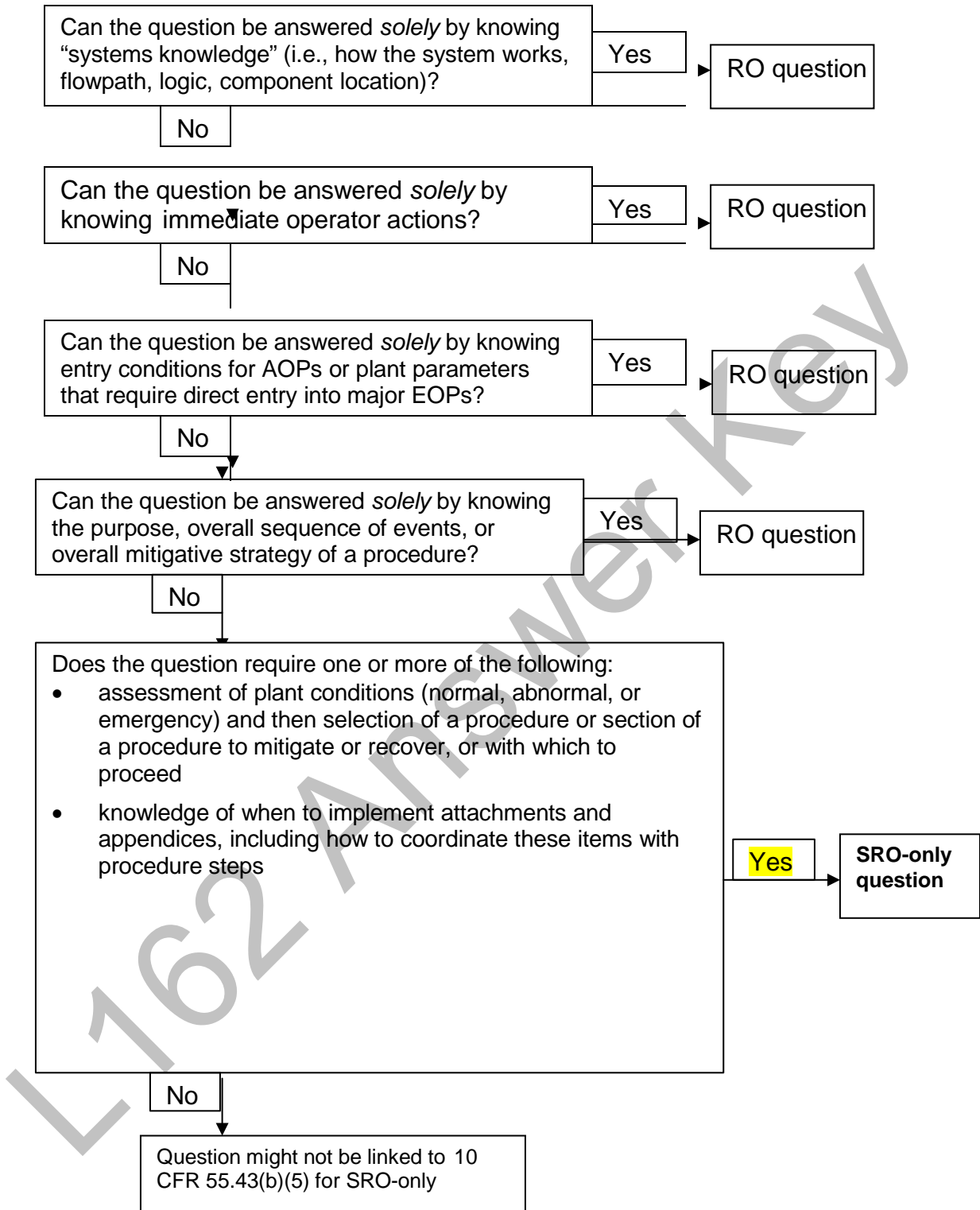
References to be provided to applicants during exam: none

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.5	

Difficulty: 2.0

SRO Justification:



Examination Outline Cross-Reference

Level	SRO
Tier #	3
Group #	3
K/A #	G2.3.6
Rating	3.8

G2.3.6 -Ability to approve release permits**Question 98**

In accordance with Form 69-21595, Gas Decay Tank Discharge Authorization, _____ 1) _____ is responsible for preparing and _____ 2) _____ is responsible for approving a Unit 1 gaseous radwaste discharge permit.

- A. 1) Chemistry 2) Shift Foreman
- B. 1) Chemistry 2) Shift Manager
- C. 1) Radiation Protection 2) Shift Foreman
- D. 1) Radiation Protection 2) Shift Manager

Proposed Answer: A. 1) Chemistry 2) Shift Foreman

Explanation:

- A. Correct. The permit is prepared by chemistry and approved by the Shift Foreman.
- B. Incorrect. The shift manager has overall control of the plant, but the SFM approves work or discharges on their unit.
- C. Incorrect. While offsite dose is part of the permit, the calculation and preparation of the permit is done by chemistry.
- D. Incorrect. Prepared by chemistry, approved by the SFM.

Technical References: OP G-2:V steps 3.2 and 3.3

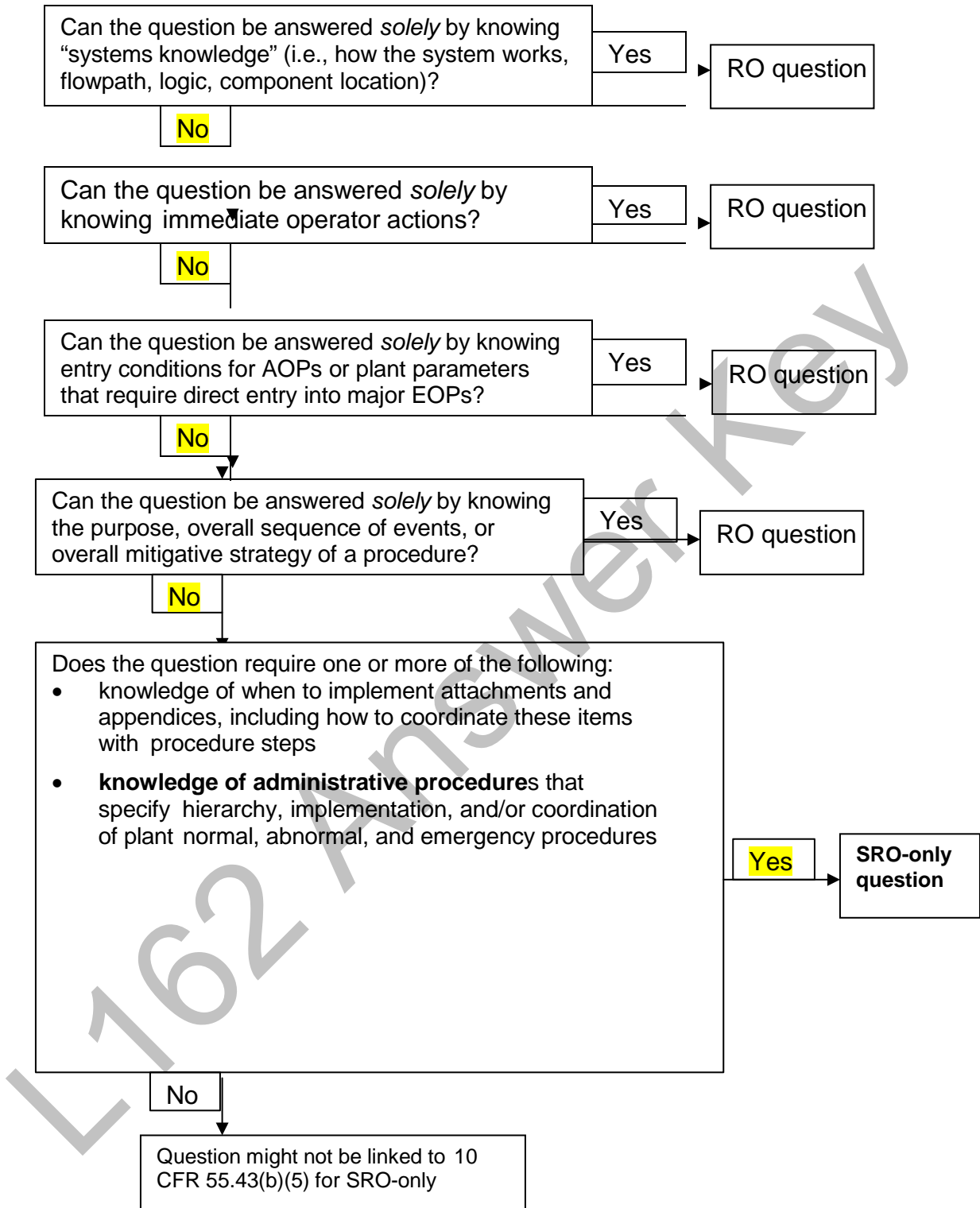
References to be provided to applicants during exam: none

Learning Objective: 8443 - State the administrative requirements of Liquid Rad Waste system

Question Source:	Bank #97 DCPN NRC 03/2012	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.4	

Difficulty: 2.5

SRO Justification:



Examination Outline Cross-Reference**G2.4.5 - Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.**

Level	SRO
Tier #	3
Group #	4
K/A #	G2.4.5
Rating	4.3

Question 99

The crew is performing FR-C.2, Response to Degraded Core Cooling.

Which of the following would require the crew to stop performing FR-C.2 prior to its completion?

- 1) RWST level of 33%
 - 2) Negative Intermediate Range Startup Rate of -0.1 DPM
 - 3) AFW flow is 400 gpm and steam generator narrow range levels are reading 0%
- A. 1 ONLY
- B. 2 ONLY
- C. Either 1 or 3
- D. Either 2 or 3

Proposed Answer: C. Either 1 or 3

Explanation:

Organization of the EOP network is to remain in the FR procedure UNLESS a higher CSF is challenged (RED- MAGENTA-YELLOW), then the procedure is stopped and the higher challenge is performed.

- A. Incorrect. This is correct, RWST level of 33% requires transition to E-1.3 (per the foldout page of FR-C.2, but is incomplete, the loss of heat sink (option 3) is a "RED" also requires terminating C.2. (note, there are procedures in the emergency response network that do not have the foldout page requirement, such as FR-S.1. Additionally, there are notes and cautions that certain procedures are not performed (such as in E-1.3 which states FR procedures are not addressed until aligned for cold leg recirc)
- B. Incorrect. This is not the MAGENTA path for subcriticality. A positive startup rate is the MAGENTA path and would then require stopping FR-C.2 because it would be a higher priority. Priority is Subcriticality, Core Cooling, Secondary Heat Sink, RCS Integrity, Containment Integrity and RCS Inventory (SCHPZI).
- C. Correct. Both the RWST level and the Heat Sink RED path would require stopping the performance of C.2.
- D. Incorrect. There is a RED path for loss of secondary heat sink, however, the challenge to subcriticality is not a magenta and does not require stopping

Technical References: F-0, FR-C.2

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: (note changes; attach parent)	Bank # Modified Bank # New Past NRC Exam Last Two NRC Exams	X No No
Question History:		
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X X
10CFR Part 55 Content:	55.43.5	
Difficulty: 2.0		

L162 Answer Key

SRO Justification:

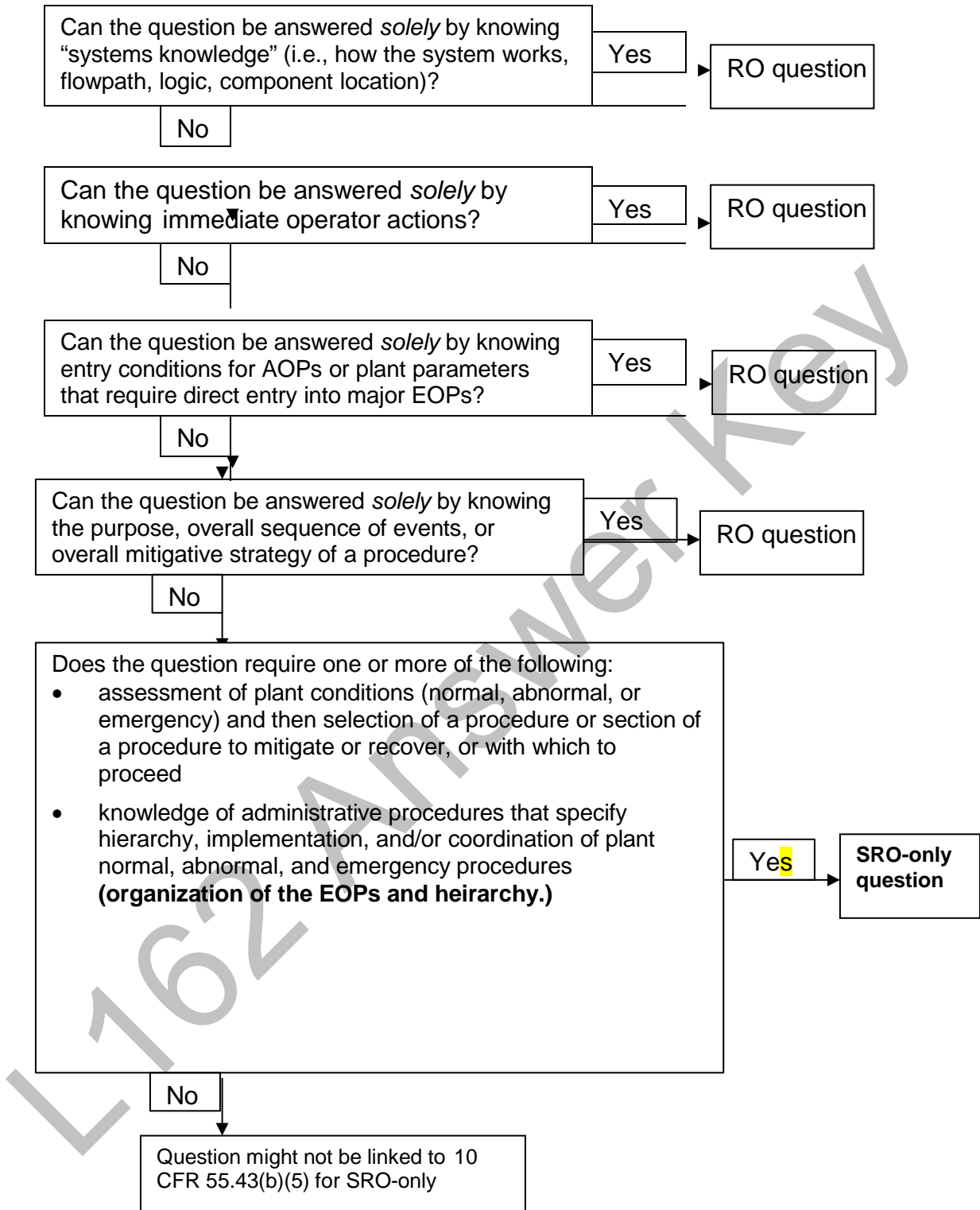
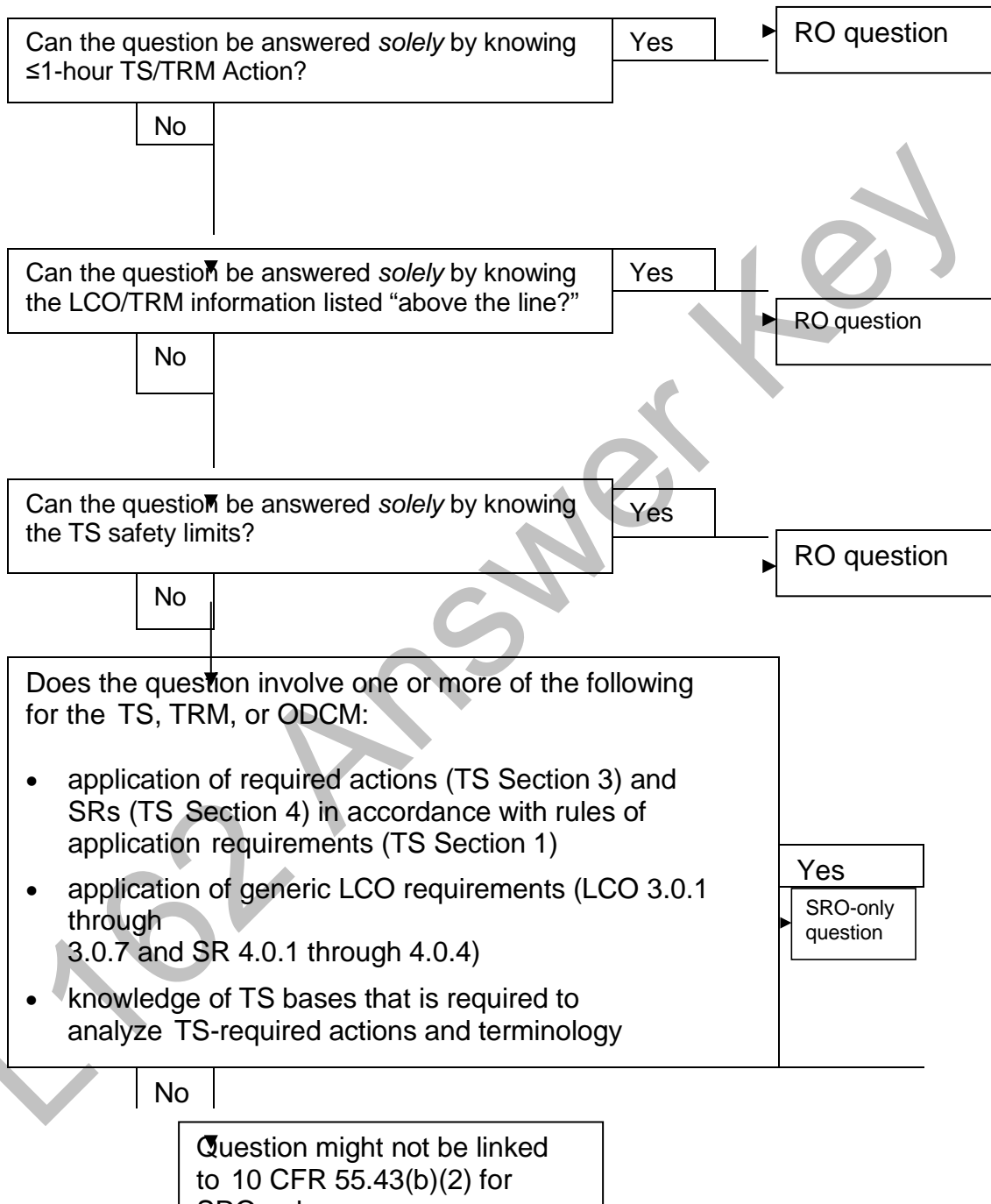


Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Examination Outline Cross-Reference

Level	SRO
Tier #	3
Group #	4
K/A #	G2.4.37
Rating	4.7

G2.4.37 - Knowledge of the lines of authority during implementation of the emergency plan.

Question 100

An ALERT is declared at 1300.

- 1) _____ may downgrade the classification.
 - 2) Both the county and state must initially be notified of the ALERT no later than _____.
- A. 1) Either the Shift Manager or the Emergency Director 2) 1315
- B. 1) Only the Emergency Director 2) 1315
- C. 1) Either the Shift Manager or the Emergency Director 2) 1400
- D. 1) Only the Emergency Director 2) 1400

Proposed Answer: B. 1) Only the Emergency Director 2) 1315

Explanation:

- A. Incorrect. The Shift Manager is the Emergency Director until relieved, which occurs after activation. The SM, as ED, may upgrade a classification but can only downgrade an Unusual Event. The county and state must be notified within 15 minutes.
- B. Correct. Per EP G-1, only the ED has the authority to downgrade a classification of alert or higher. 15 minutes to notify State and County.
- C. Incorrect. The SM can only downgrade a NUE. This is the time when NRC notification is required.
- D. Incorrect. Only the ED can downgrade the ALERT and the NRC must be notified within an hour of classification, but 15 minutes is what is required for state and county.

Technical References: EP G-1, G-3

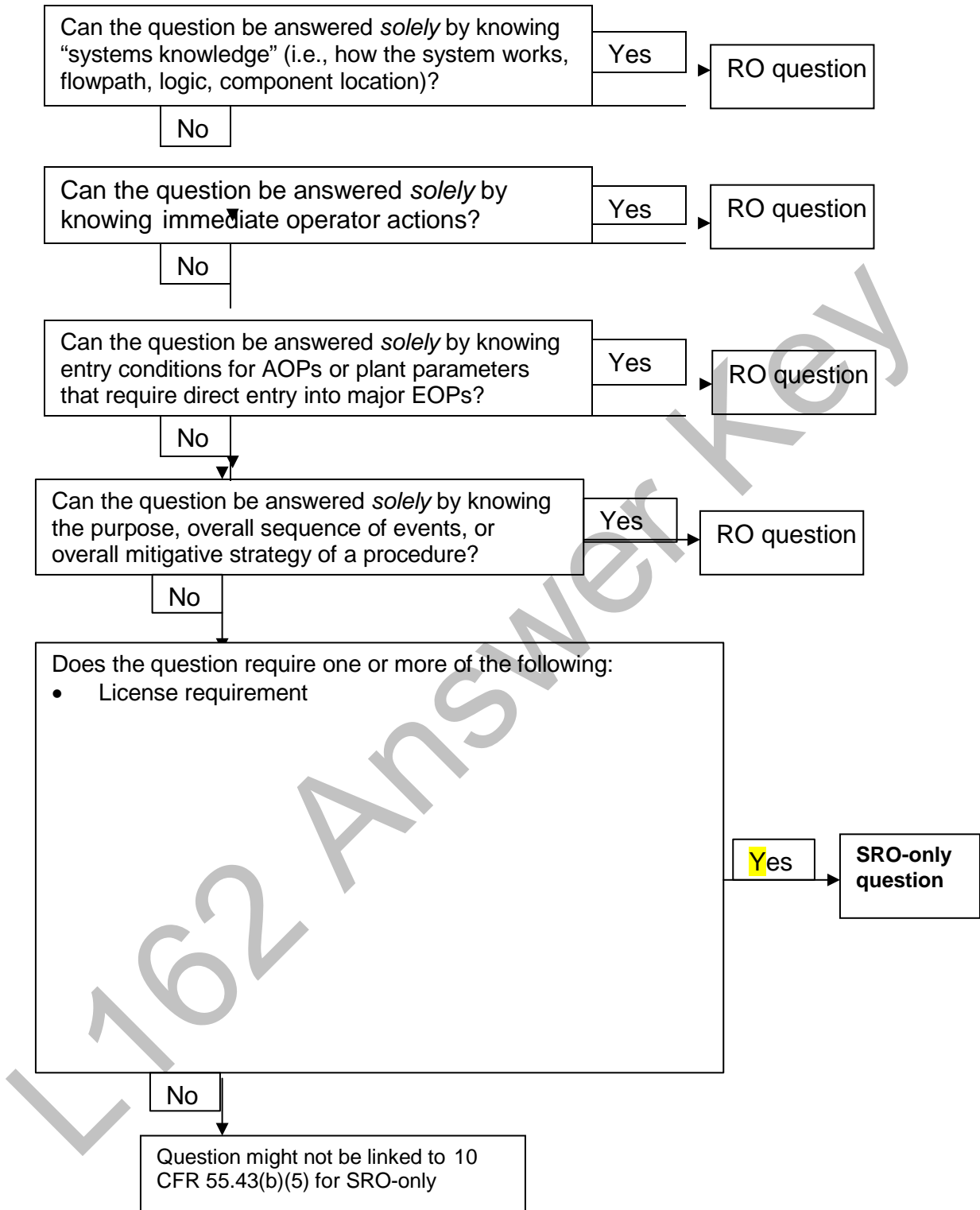
References to be provided to applicants during exam: none

Learning Objective: 42287 -As described in EP G-3, state the requirements for the following: Initial Notifications (Time Critical)

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

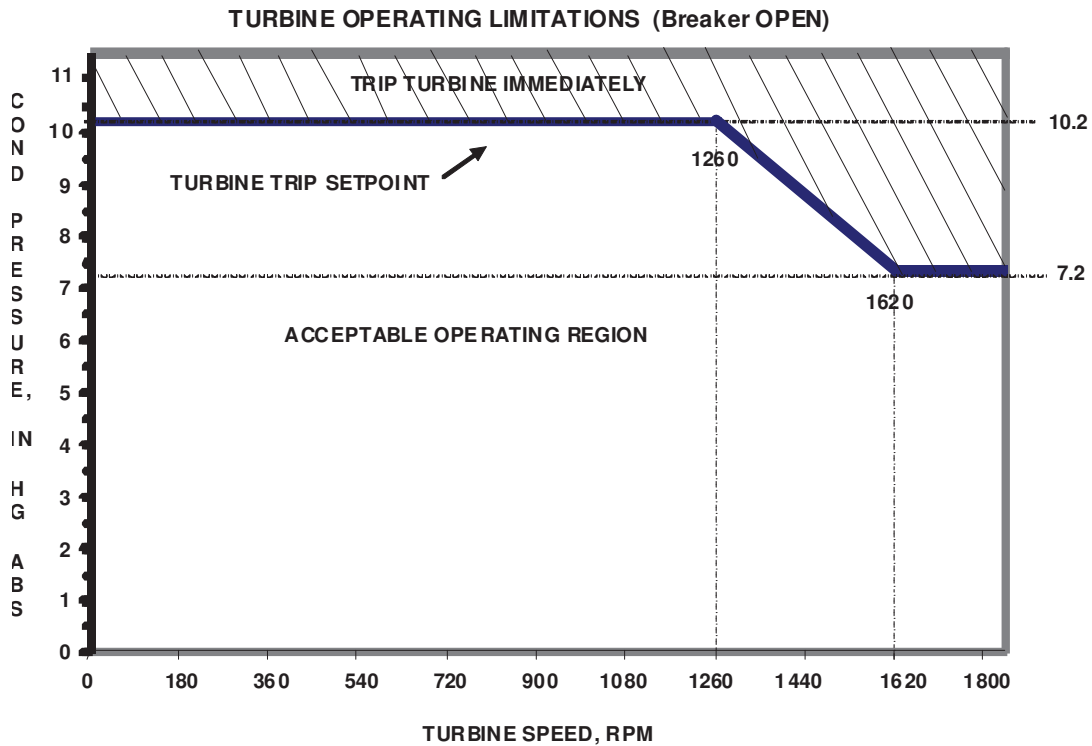
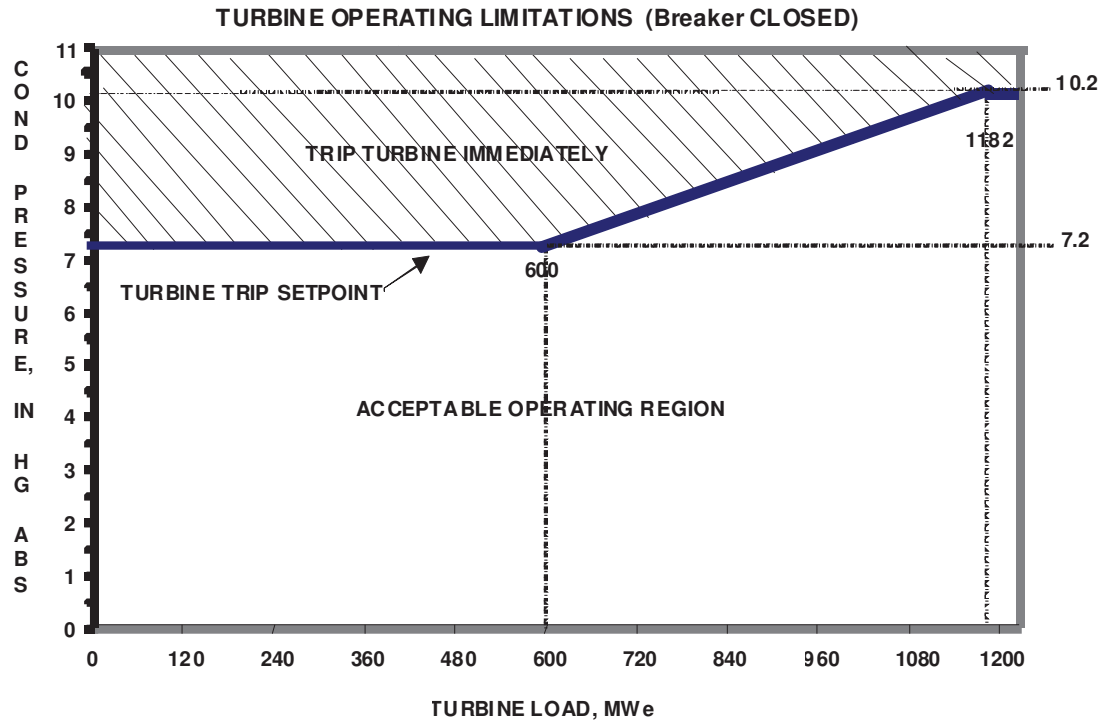
Difficulty: 2.5

SRO Justification:



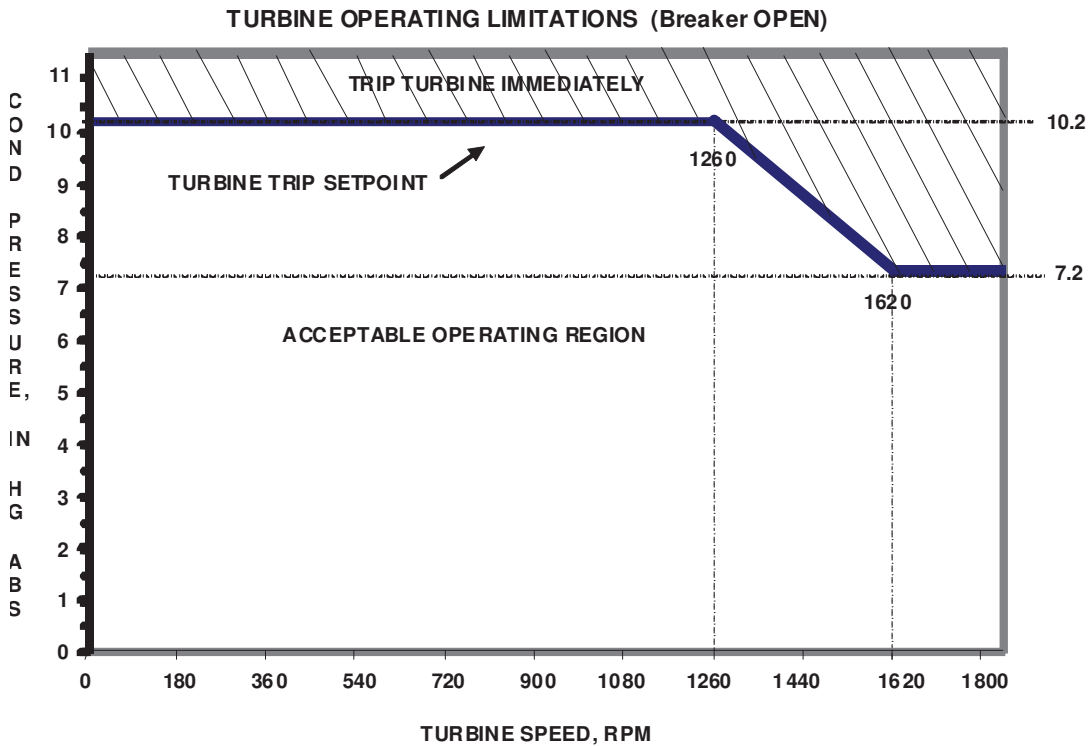
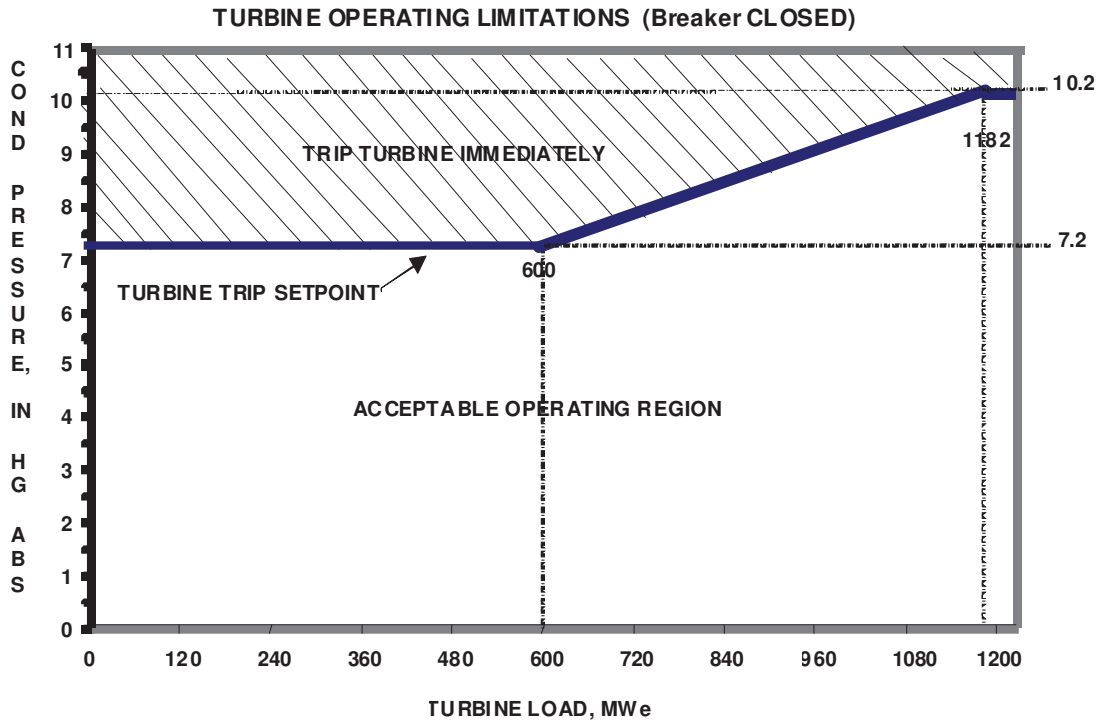
Turbine Operating Limitations

U1&2 Attachment 2: Page 1 of 1



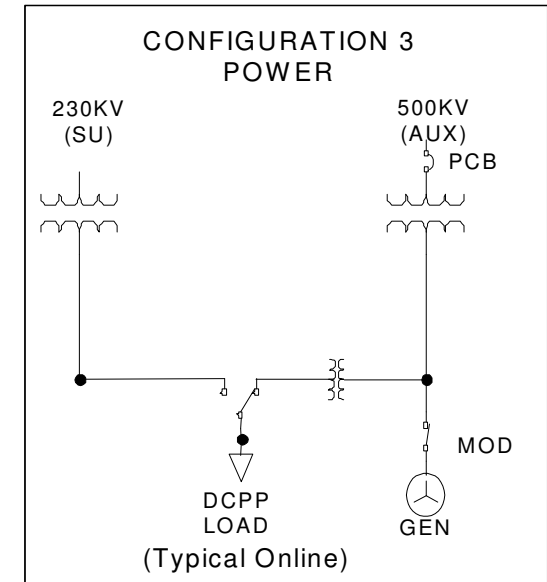
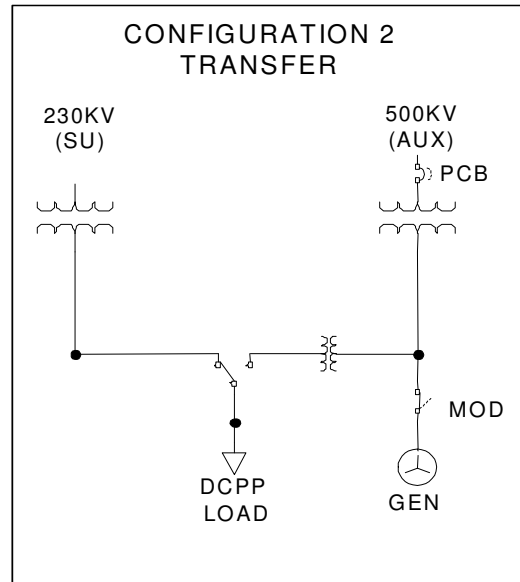
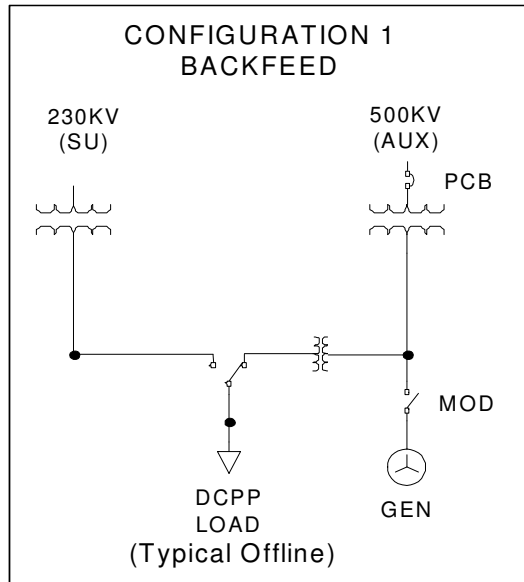
Turbine Operating Limitations

U1&2 Attachment 2: Page 1 of 1



Diablo Canyon Electrical Alignment Configurations

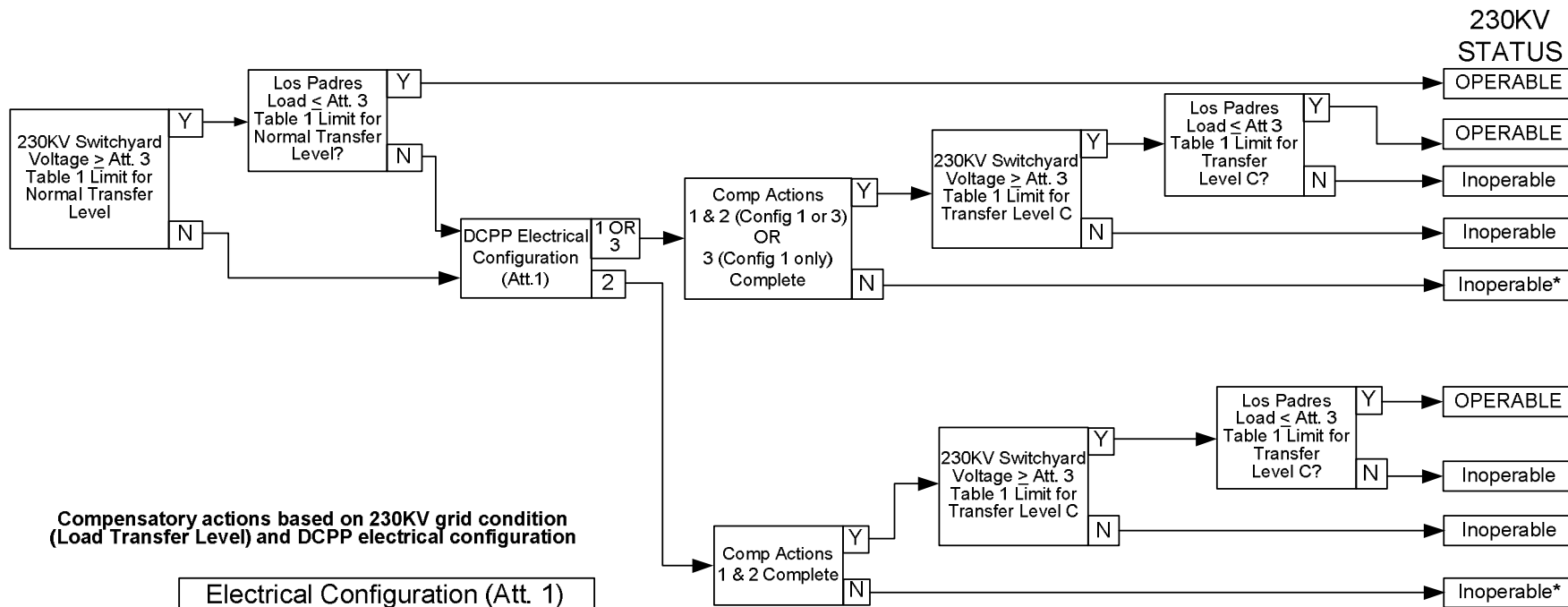
U1&2 Attachment 1: Page 1 of 1



NOTE: Configuration 2 (Transfer) applies when at least one 12kV bus is aligned to Startup; other buses aligned to either Startup or Aux.

230kV Offsite Power Operability Evaluation - LTC IN AUTO

U1&2 Attachment 2: Page 1 of 1



Compensatory actions based on 230KV grid condition (Load Transfer Level) and DCPP electrical configuration

		Electrical Configuration (Att. 1)		
		Config 1 Backfeed	Config 2 Transfer	Config 3 Power
Load Transfer Level (Att. 3 Table 1)	Norm	None	None	None
	A	Levels A & B Not Used		
	B			
	C	(1 & 2) OR 3	1 & 2	1 & 2

* - Determine if performing compensatory actions will restore operability.

- Compensatory Actions**
1. Auto start on unit trip of ALL standby Condensate/Booster Pump Sets blocked.
 2. Auto transfer of 12KV Bus D **OR** Bus E blocked.
 3. Both 12KV Buses D AND E unloaded.



Operating Instruction for Reliable Transmission Service to Diablo Canyon Power Plant

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IMPORTANT: Contact DCP" within 10 minutes when real-time system conditions move to new row.																																																																																																																
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Operating Instruction for Reliable Transmission Service to Diablo Canyon Power Plant

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O23 Rev	4.0																																							
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<p>1 Both DCP" Units Off-line (i.e. no generation)</p> <p>2 DCP" 230kV Baseline/Pre-trip Load:</p> <table style="margin-left: 40px;"> <tr> <td></td> <td style="text-align: center;">Light Load</td> <td style="text-align: center;">V= 230kV</td> </tr> <tr> <td style="padding-left: 20px;">MW=</td> <td style="text-align: center;">4.0</td> <td></td> </tr> <tr> <td style="padding-left: 20px;">MVAR=</td> <td style="text-align: center;">3.5</td> <td></td> </tr> </table> <p>3 Grid Conditions prior to adding DCP" post-trip incremental load</p> <p>4 DCP" 230kV Post Trip Levels</p> <table style="margin-left: 40px; border: 1px solid black; width: 80%;"> <thead> <tr> <th style="text-align: left;">Incremental Load added to grid:</th> <th style="text-align: center;"><u>Level</u></th> <th style="text-align: center;"><u>MW</u></th> <th style="text-align: center;"><u>MVAR</u></th> <th style="text-align: center;"><u>MVA</u></th> <th style="text-align: center;"><u>Pf</u></th> </tr> </thead> <tbody> <tr> <td></td> <td style="text-align: center;">Normal</td> <td style="text-align: center;">90.0</td> <td style="text-align: center;">67.5</td> <td style="text-align: center;">113.0</td> <td style="text-align: center;">0.80</td> </tr> <tr> <td></td> <td style="text-align: center;">Level A</td> <td style="text-align: center;">---</td> <td style="text-align: center;">---</td> <td style="text-align: center;">---</td> <td style="text-align: center;">---</td> </tr> <tr> <td></td> <td style="text-align: center;">Level B</td> <td style="text-align: center;">---</td> <td style="text-align: center;">---</td> <td style="text-align: center;">---</td> <td style="text-align: center;">---</td> </tr> <tr> <td></td> <td style="text-align: center;">Level C</td> <td style="text-align: center;">70.0</td> <td style="text-align: center;">52.5</td> <td style="text-align: center;">88.0</td> <td style="text-align: center;">0.80</td> </tr> </tbody> </table> <p>5 Grid Operating Limits to ensure at least 207kV post-trip voltage on local 230kV system and not exceed DCP" NPIRs</p> <p>6 For transmission operating conditions or configurations other than those described in this table, the utility Operations Engineering Dept and DCP" Design Engineering Dept should coordinate and confirm new/updated limits.</p> <p>7 Used when any line section of the #1 circuit of this Morro Bay-Solar SS-Caliente SS-Midway path is Out Of Service (OOS)</p> <p>8 Used when any line section of the #2 circuit of this Morro Bay-Solar SS-Caliente SS-Midway path is Out of Service (OOS).</p> <p>9 Used when this path has the Out Of Service section closest to Morro Bay during multiple (non-adjacent) line outages</p> <p>10 IN = Capacitor manually inserted or under control of AVC (Automatic Voltage Controller)</p> <p>11 Used when EITHER line section of the Morro Bay-Gates 230kV path is Out of Service (OOS).</p>			Light Load	V= 230kV	MW=	4.0		MVAR=	3.5		Incremental Load added to grid:	<u>Level</u>	<u>MW</u>	<u>MVAR</u>	<u>MVA</u>	<u>Pf</u>		Normal	90.0	67.5	113.0	0.80		Level A	---	---	---	---		Level B	---	---	---	---		Level C	70.0	52.5	88.0	0.80
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